Safety Reports Series

No. 123

Applicability of IAEA Safety Standards to Non-Water Cooled Reactors and 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 on the IAEA Internet site

https://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.

APPLICABILITY OF IAEA SAFETY STANDARDS TO NON-WATER COOLED REACTORS AND SMALL MODULAR 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
CABO VERDE
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
GAMBIA
GEORGIA
GERMANY
GHANA
GREECE
GRENADA
GUATEMALA
GUINEA
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".

SAFETY REPORTS SERIES No. 123

APPLICABILITY OF IAEA SAFETY STANDARDS TO NON-WATER COOLED REACTORS AND SMALL MODULAR REACTORS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2023

Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 26007 22529
tel.: +43 1 2600 22417
email: sales.publications@iaea.org
www.iaea.org/publications

Printed by the IAEA in Austria
November 2023
STI/PUB/2027

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Applicability of IAEA safety standards to non-water cooled reactors and small modular reactors / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2023. | Series: IAEA Safety Reports Series, ISSN 1020-6450 ; no. 123 | Includes bibliographical references.
Identifiers: IAEAL 23-01621 | ISBN 978–92–0–127323–9 (paperback : alk. paper) | 978–92–0–127423–6 (pdf) | ISBN 978–92–0–127523–3 (epub)
Subjects: LCSH: Nuclear reactors — Safety measures. | Nuclear reactors
Classification: UDC 621.039.56 | STI/PUB/2027

FOREWORD

In recent years, there has been growing interest among Member States in the development and deployment of non-water cooled reactors and small modular reactors. These types of reactor may use innovative safety technologies, including passive and inherent safety features, various types of fuel and coolant, and various approaches to practically all aspects of a reactor lifetime, such as construction, operation, waste management, decommissioning and transportation. Therefore, non-water cooled reactors and small modular reactors present important areas of novelty compared with the current fleet of large, land based water cooled reactors.

As with the current fleet of reactors, non-water cooled reactors and small modular reactors must meet the objective of protecting people and the environment and minimizing the possibility of accidents. A key element of meeting this objective is demonstrating compliance with the IAEA safety standards. The safety standards reflect consensus among Member States and cover a wide range of aspects relevant to the lifetime of nuclear facilities, such as regulation, siting, design, construction, commissioning, operation, decommissioning, release from regulatory control and radioactive waste management. The IAEA safety standards have been largely informed by the experience and knowledge of Member States on the current fleet of reactors.

In view of the increase in global activities related to non-water cooled reactors and small modular reactors, this Safety Report documents the areas of novelty of these technologies in comparison with the existing fleet of reactors and provides an assessment of their impact on the applicability and completeness of the IAEA safety standards. It also provides an overall review of the extent to which the current safety standards address the safety of non-water cooled reactors and small modular reactors. This review includes identifying any gaps (i.e. areas of novelty not covered by the safety standards) and areas for additional consideration, and its scope encompasses all the safety standards related to the lifetime of these reactor technologies, as well as the interfaces between safety, security and safeguards in their design.

The broad scope of this publication makes it valuable to regulatory bodies, technical support organizations, operating organizations of nuclear power plants, vendor companies (e.g. designers, engineering contractors, manufacturers) and research establishments. The applicability review presented in this Safety Report was conducted at a high level, and the development of a more detailed assessment may be advisable, especially for areas of high relevance to safety or for which gaps or areas for additional consideration were identified.

The IAEA is grateful to the many experts from Member States who contributed to this applicability review. The IAEA officers responsible for

this publication were P. Calle Vives and G. Martinez-Guridi of the Division of Nuclear Installation Safety.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The IAEA safety standards are based on good practices drawn from the experience of Member States with nuclear and radiation related technology intended for peaceful purposes. In the case of nuclear power plants (NPPs), much of this experience relates to large, land based water cooled reactors (WCRs) that are dedicated to electricity generation. Although safety standards aim to be technology neutral, their content does sometimes reflect the current dominance of this type of NPP.[1] Technology is rarely static, however, and Member States have expressed significant interest in commercial deployment of other types of reactor, notably non-WCRs and small modular reactors (SMRs; water cooled or non-water cooled). These technologies may be large or small, transportable or not, land based or marine based; they may be developed for applications other than electricity production (e.g. heat generation); and they may currently be in very different stages of development, licensing and deployment. These reactor technologies are referred to in this publication as ‘evolutionary and innovative designs’ (EIDs) for the sake of brevity and following the terminology in Ref. [1].

Compared with the currently operating WCRs, EIDs present new features and innovative technologies, including different types of coolant, nuclear fuel, neutron spectra and inherent safety features, as well as concepts based on modularity (such as SMRs). Their emergence also impacts support activities such as component manufacture, fuel storage, spent fuel reprocessing, predisposal management of waste and waste disposal. There is a wide range of EIDs for immediate, near term and medium term deployment worldwide. For example, Ref. [2], which is not comprehensive, identifies over 50 different SMR designs. As prescribed by national regulations, all reactors, including EIDs, have to be operated in a safe manner to protect people and the environment from the harmful effects of ionizing radiation. A key element of demonstrating that this objective has and will be met is compliance with the IAEA’s fundamental safety principles and safety requirements.

IAEA safety standards, which represent an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation, comprise an extensive documentation to support nuclear safety by establishing safety principles, requirements,

[1] Experience with non-WCR technologies also exists from the small number of past and existing facilities but was not explicitly addressed in the requirements and guidance of the current safety standards.

and associated recommendations and guidance. IAEA safety standards have been developed in an iterative fashion; they cover all aspects of the peaceful exploitation of nuclear technology, from early governmental decisions through to radioactive waste disposal, with regulation for safety being a recurring theme throughout.

The current growing interest among the Member States in the design, development and deployment of EIDs makes it necessary to consider the applicability of the existing IAEA safety standards to these technologies and to identify any gaps that might exist. It is also important to identify additional considerations that may exist when considering the interface between safety, security and safeguards considerations (referred to in this publication as the '3S concept').

The IAEA has undertaken various initiatives to review the applicability of certain safety standards to some types of non-WCR and some types of SMR. Furthermore, the SMR Regulators' Forum has evaluated the applicability of some IAEA safety standards to SMRs and has made recommendations for additional guidance to be developed by the IAEA. Until now, however, the IAEA had not conducted a review or reported an overview of the application of relevant safety standards to the entire lifetimes of EIDs, including SMRs. This publication documents such a review and the findings on the applicability of these standards to different types of EID. Its main goal is to improve understanding on the topic and to provide a basis for a comprehensive and coordinated approach to updating the current standards, developing new standards and applying the standards in practice.

1.2. OBJECTIVE

The objective of this publication is to present a high level review of the applicability of IAEA safety standards to EIDs (including SMRs) — in particular, to consider whether the current requirements and recommendations are applicable to these technologies and to identify any gaps, that is, new safety issues on which the standards appear to be silent or not fully address. The publication also identifies specific considerations related to the interfaces between safety, security and safeguards.

The publication is intended for use by regulatory bodies, technical support organizations, operating organizations of nuclear power plants, vendor companies (e.g. designers, engineering contractors, manufacturers) and research establishments.

Guidance and recommendations provided here in relation to identified good practices represent expert opinion but are not made on the basis of a consensus of all Member States.

1.3. SCOPE

This publication is based on the results of a high level review of the applicability of the IAEA safety standards to EIDs and their associated facilities, such as those for fuel fabrication and spent fuel management. The review considered the entire lifetime of these technologies (i.e. the stages of siting, design, construction, commissioning, operation and decommissioning, including related nuclear fuel cycle facilities, radioactive waste management, safety assessment, emergency preparedness and response, and transport).

This publication also considers the interfaces between safety, security and safeguards for EIDs. The areas of novelty for EID technologies are examined from the perspectives of security and safeguards to determine their impact on current practices in these areas. However, a detailed analysis of the applicability of IAEA security and safeguards publications to EIDs is outside the scope of this publication.

Regarding molten salt reactors (MSRs), only the applicability of IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [3] was considered, and the applicability of other IAEA safety requirements was not included at this time. However, this publication includes some insights on MSRs from a 3S perspective.

Evaluations presented in this publication concerning transportable NPPs (TNPPs) and microreactors are regarded as preliminary, given that these items were introduced to the work programme late in the process.

1.4. STRUCTURE

This publication comprises six sections and two appendices. Section 2 presents the approach to the high level review of the applicability of the IAEA safety standards to EIDs and their associated facilities, as well as an overview of the IAEA safety standards that were considered in the review, grouped broadly according to the relevant stages in the lifetime of a nuclear facility, such as siting, design, construction, operation and decommissioning.

Section 3 identifies areas of novelty at the various stages in the lifetime of an EID for the technologies presented in Section 2.1.2.

Section 4 lists individual safety standards and examines them in terms of whether (i) the standard may be applicable to EIDs, (ii) the standard may not be applicable, owing to the specificities of the technologies, or (iii) the standard may not cover some of the areas of novelty of EIDs.

Section 5 examines the potential for additional considerations regarding the interfaces between safety, security and safeguards in relation to the areas of novelty. Appendix I presents a summary of additional considerations and challenges related to the IAEA safeguards of EIDs.

Section 6 gives the key outcomes of the applicability review of safety standards to EIDs. Appendix II presents tables summarizing these outcomes.

2. IAEA APPROACH AND SAFETY STANDARDS

2.1. APPROACH

2.1.1. General

The applicability of IAEA safety standards to EIDs and their associated technologies was reviewed because these safety standards were developed when WCRs were pre-eminent. The review approach followed the main steps listed below:

(a) Identification of areas of novelty of EIDs compared with WCRs. This identification is based on a systematic comparison of the characteristics of EIDs with a WCR reference case. The characterization was initially developed on the basis of expert knowledge, literature review and responses by technology developers to detailed questionnaires prepared by the IAEA. This characterization was then reviewed by regulatory authorities, technical support organizations and other organizations from Member States participating in this project. The characterization is summarized in two working documents (unpublished) and the literature review included numerous publications.

(b) Comparison of the identified areas of novelty with the requirements and recommendations in the IAEA safety standards to identify areas where the standards (i) may be applicable; (ii) may not be applicable; and (iii) may have gaps or, in some cases, may require further work to become applicable to EIDs or to a subset of EIDs. Such cases include known technology specific areas for which the application of the standards to EIDs requires additional guidance.

(c) Consideration of the areas of novelty from the security and safeguards perspectives to identify any additional considerations related to the interface between safety, security and safeguards.

The term ‘gap’ is used very frequently in this publication, especially in Section 4, to denote an aspect of a type of EID that the IAEA safety standards do not fully cover. This is an indication that the current standards would not be able to fully address this specific aspect. This publication does not attempt to assess the significance of gaps — that is, whether they are important or not — nor does it suggest possible solutions.

2.1.2. Technologies considered

For the purpose of this publication, all the technologies presented in this section are considered as EIDs. While some of these technologies have already been deployed, they are significantly different from the technology of WCRs with which practitioners are most familiar. This familiarity has, intentionally or not, influenced the content of the IAEA safety standards.

Common characteristics of the technologies covered by this publication is the limited available operating experience and the even less commercial operating experience. Some EIDs have not yet been commercially operated and many EIDs are still to be licensed and deployed in a Member State.

The study includes EIDs of different technologies, presented in Sections 2.1.2.1–2.1.2.6. These have been grouped and defined following the information in the Advanced Reactors Information System (ARIS) database [4] and the supplementary information in Ref. [2], complemented by questionnaire responses from EID developers and vendors and, for non-WCRs, by the Generation IV International Forum (GIF) [5]. Where new technologies have been applied to WCRs, these new reactor types are considered within the definition of the water cooled reactor reference plant; they include various types of pressurized water reactor (PWR), boiling water reactor (BWR) and pressurized heavy water reactor (PHWR). Reactor concepts that are likely inactive and design studies were not included.

2.1.2.1. *Water cooled small modular reactors*

The IAEA database ARIS [4] and the supplementary information related to SMRs in Ref. [2] present three main WCR types:

(1) SMRs with a BWR type reactor;
(2) Integral PWRs;
(3) Other water cooled SMRs — either heavy water cooled and moderated SMRs or water cooled SMRs that do not fall into WCR types 1 and 2.

2.1.2.2. *Sodium cooled fast reactors*

Designs of sodium cooled fast reactors (SFRs) that are considered in this publication include loop types[2] and pool types[3]. All of them use sodium in the primary system, while the secondary (intermediate) coolant system may use sodium or molten salt as coolant. The transferred heat is then used to drive a turbine alternator via a conventional water/steam cycle, supercritical CO_2 power cycle, or similar. Some SFR designs may be equipped with a molten salt heat storage system, which may be used for various applications such as desalination, steam production and other industrial applications. A range of fuel types includes oxide (UO_2 and MOX), metal, nitride, carbide and minor actinide-bearing fuels.

SFRs have accumulated development, operation and decommissioning experience, and have reached demonstration and commercial phases in some countries. In parallel, new SFR designs are being developed (e.g. small modular SFRs) by some private companies that are newly involved in SFR technology.

2.1.2.3. *Lead cooled fast reactors*

Lead cooled fast reactors (LFRs) included in this review are of the pool type. All of them use lead (in some cases lead–bismuth) as the primary coolant system, with water, supercritical water or supercritical CO_2 as secondary coolant used to drive a turbine alternator. One design opts for three circuits by including a lead filled secondary circuit. The fuel types considered are metal fuel, UO_2, MOX, mixed nitrides and minor actinide-bearing fuels.

---

[2] A loop type design has primary system components (i.e. steam generator, primary coolant pump, pressurizer) connected to each other and to the reactor vessel through large pipes. The design may have more than one loop.

[3] In a pool type design, the reactor core is immersed in a pool of coolant.

2.1.2.4. High temperature gas cooled reactors

The study includes high temperature gas cooled reactors (HTGRs) of both pebble bed and prismatic block designs. All of them use helium in the primary coolant system, which is then coupled to a secondary heat exchange system that uses water/steam, helium, nitrogen, a mix of nitrogen and helium, or molten salt; the choice is mainly decided by the intended application. All of these reactors use TRISO (tristructural isotropic) fuel.

2.1.2.5. Molten salt reactors

Many MSR concepts are being considered in section 5 of Ref. [6]. Some MSR developers propose commercial deployment within the next 10 years, while other designs are still being developed, making it likely that new MSR concepts will be proposed in the future. This study considers the following two main MSR types:

(1) Solid fuel MSRs, in which molten salt is used only as the coolant;
(2) Liquid fuel MSRs, in which the fissile material is dissolved in molten salt, which circulates through a heat exchanger so that the molten salt acts as both fuel and coolant.

These types of MSR include reactors with thermal, epithermal and fast neutron spectra. MSRs with a thermalized neutron spectrum use graphite as moderator. Some MSR designs may be equipped with a molten salt heat storage system that, as described above for SFR, brings added flexibility.

2.1.2.6. Transportable nuclear power plants

A TNPP is considered herein to be a feature of some EIDs, and not a technology per se. It is included here because some TNPPs are already in operation, and there are plans for further deployment by some Member States, particularly for SMR designs. In addition, they introduce specific challenges for design, transportation, operation and safety assessment that may not be reflected in current IAEA guidance. A TNPP consists of a nuclear power plant that is designed to be geographically relocated as a complete, or near complete, system. Typically, TNPPs have a long fuel dwell time to reduce the frequency of refuelling. Very small TNPPs (less than 10 MW(e)) are often called microreactors. While TNPPs are physically transportable, they are not designed to either produce energy during transportation or provide energy for the transportation itself. TNPPs are often intended for electricity generation in remote areas, for district heating, desalination of sea water and hydrogen production. As previously indicated, this publication includes only preliminary consideration of TNPPs.

2.1.3. Clarifications on terminology

Because many EIDs are still under development, designs are often not complete, meaning that their potential vendors may have made claims for their EIDs that have yet to be substantiated. In the absence of more detailed information, the experts that contributed to this study have made statements such as “may be less stringent”, “may not be needed ”, “may be acceptable”, and “may not require a safety class 1”. Readers should be aware that such statements, much less the publication as a whole, must not be construed as endorsing these claims.

An NPP is generally understood to be a power plant with a single reactor unit on a site, and most IAEA safety standards are based on this definition. There may be several NPPs (each with a single reactor unit) at a site. This term is used herein in a broad sense, to include multiple identical reactor modules as part of an SMR; the SMR, with its various reactor modules, is an NPP. Hence, there may be several reactors at a site.

The term ‘reactor module’ refers to a “nuclear reactor with its associated structures, systems and components” [7].

Every type of reactor mentioned in Sections 2.1.2.1–2.1.2.6 represents a kind of NPP that has one specific type of reactor. For example, the term SFR means an NPP with this type of reactor, and SMR means an NPP having one or more reactor modules (typically between 20 and 300 MW(e)) of the same technology, such as SFRs.

2.2. STRUCTURE OF THE IAEA SAFETY STANDARDS SERIES

The structure (hierarchy) of the IAEA Safety Standards Series comprises three categories (Fig. 1). At the top (first tier) are the Safety Fundamentals, which present the fundamental safety principles. In the middle (second tier) are the Safety Requirements, which establish the requirements that need to be met to satisfy the fundamental safety principles. In the third tier are the Safety Guides, which advise Member States on how to comply with the safety requirements. The level of detail and technology specificity increases in the lower levels of this hierarchy (Safety Requirements and Safety Guides).

All General Safety Requirements publications (GSR Part 1 (Rev. 1) to GSR Part 7) [8–14] have been developed with the intention that they may be applicable to any kind of activity or facility, independently of type and their associated technologies.

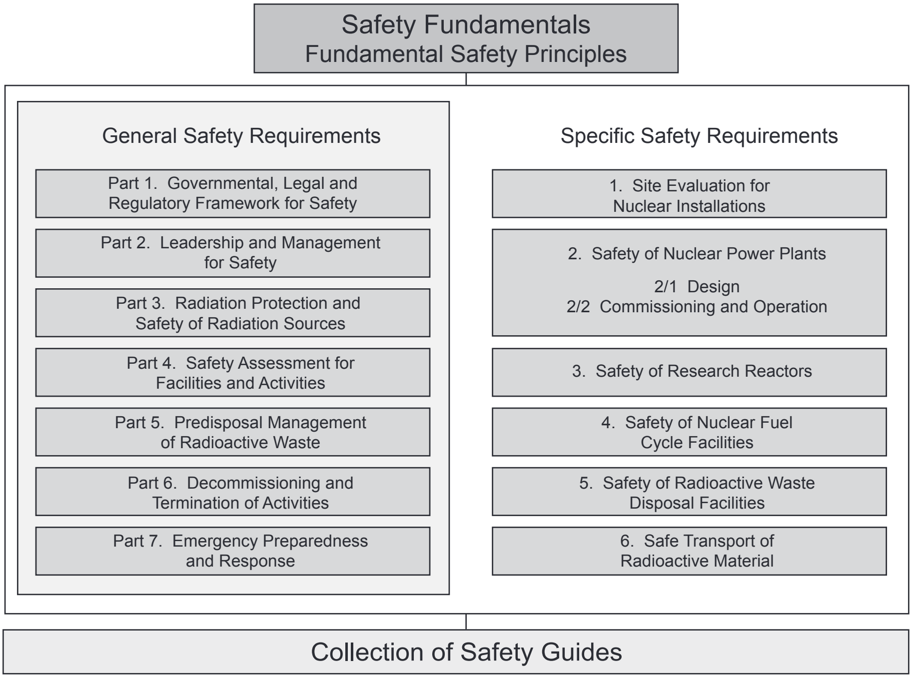


FIG. 1. Structure of the IAEA Safety Standards Series.

Similarly, all General Safety Guides [15–20] contain guidance that has been developed with the rationale that they may be applicable to any kind of activity or facility and their associated technologies. Consequently, there is confidence that the General Safety Requirement and General Safety Guide publications are non-specific with respect to technology.

Specific Safety Standards publications may contain requirements or guidance that may not be technology neutral.

This publication presents a systematic review of all the safety requirements and guidance (i.e. including both general and specific) that are within the scope of this study and identifies areas of EIDs where the standards (a) are directly applicable; (b) may not be directly applicable; and (c) may not provide adequate coverage. The overall aim is to evaluate the applicability of the IAEA safety standards to EIDs.

2.3. SAFETY REQUIREMENTS CONSIDERED

This section provides a high level description of the requirements for NPPs.

2.3.1. Siting of nuclear installations

IAEA Safety Standards Series No. SSR-1, Site Evaluation for Nuclear Installations [18], establishes the requirements for the elements of a site evaluation for an NPP (and, in general, for a nuclear installation) to fully characterize the site specific conditions pertinent to the safety of the NPP.

2.3.2. Design and construction

IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [3], presents 82 requirements in relation to the design of land based stationary NPPs with WCRs. Even though SSR-2/1 (Rev. 1) does not make a distinction on the size of the reactor, it was likely influenced by the experience with large NPPs. SSR-2/1 (Rev. 1) states that it may also be applied, with judgement, to other reactor types, establishes a requirement on construction (Requirement 11) and considers the interface between design and construction as part of several design requirements.

2.3.3. Commissioning and operation

IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation [19], aims to "establish the requirements which, in the light of experience and the present state of technology, must be satisfied to ensure the safe commissioning and operation of nuclear power plants." Since nuclear safety should be factored into all operating decisions, the requirements contained in SSR-2/2 (Rev. 1) [19] include management responsibilities, such as those that relate to appropriate staffing and training of personnel.

2.3.4. Nuclear fuel cycle facilities

The safety of nuclear fuel cycle facilities is covered by IAEA Safety Standards Series No. SSR-4, Safety of Nuclear Fuel Cycle Facilities [20]. This publication presents requirements for site evaluation, design, construction, commissioning, operation and preparation for decommissioning, all of which must be satisfied to provide an adequate level of safety. In the present context, the relevance to novel technologies will mainly be due to the introduction of new types of fuel.

2.3.5. Radiation protection and safety

IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [10], also known as the International Basic Safety Standards or simply 'the BSS', applies to all facilities and all activities that give rise to radiation risks[4]. It addresses the protection of workers, patients, the public and the environment in all exposure situations: planned, emergency and existing. GSR Part 3 [10] is jointly sponsored by eight international organizations and takes account of the findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of the International Commission on Radiological Protection (ICRP).

2.3.6. Radioactive waste management and decommissioning

IAEA Safety Standards Series No. GSR Part 5, Predisposal Management of Radioactive Waste [12], sets out the objectives, criteria and requirements for the protection of human health and the environment that apply to the siting, design, construction, commissioning, operation and shutdown of facilities for the predisposal management of radioactive waste, as well as the requirements that must be met to provide adequate levels of safety for such facilities and activities. IAEA Safety Standards Series No. SSR-5, Disposal of Radioactive Waste [21], sets out the safety objective and criteria for the disposal of all types of radioactive waste and establishes, based on the principles stated in the Safety Fundamentals, the requirements that must be satisfied in the disposal of radioactive waste.

IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [13] establishes the safety requirements for all aspects of decommissioning, from the siting and design of a facility to the termination of the authorization for decommissioning. It applies to the decommissioning of nuclear power plants, research reactors and other nuclear fuel cycle facilities, including predisposal waste management facilities.

2.3.7. Leadership and management for safety

Fundamental Safety Principle 3, as established in IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [22], calls for effective leadership and management for safety in order to establish and apply requirements for

---

[4] In the IAEA Safety Standards, the term 'protection and safety' refers to the protection of people against exposure to ionizing radiation or exposure due to radioactive material and the safety of sources. It does not include non-radiation related aspects of safety.

safety and promote a safety culture, among others. IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [9], establishes the relevant safety requirements as they apply to nuclear installations, registrants and licensees throughout the installation's lifetime until its release from regulatory control. These requirements also "apply in relation to the functions and activities of the regulatory body, as far as is appropriate" [9]. Requirements include the fostering of a strong nuclear safety culture, an active assessment programme to include the integration of the management system to ensure that safety is considered by all elements of management and continuous learning for all members of the operating organization.

2.3.8. Safety assessment of nuclear installations

IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [11], states:

> "Safety assessment plays an important role throughout the lifetime of the facility or activity whenever decisions on safety issues are made by the designers, the constructors, the manufacturers, the operating organization or the regulatory body. The initial development and use of the safety assessment provides the framework for the acquisition of the necessary information to demonstrate compliance with the relevant safety requirements, and for the development and maintenance of the safety assessment over the lifetime of the facility or activity."

GSR Part 4 (Rev. 1) [11] establishes the generally applicable requirements to be fulfilled in the safety assessment of an NPP (and of any other nuclear facility).

2.3.9. Emergency preparedness and response

IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [14], establishes the safety requirements for an adequate level of preparedness and response for a nuclear or radiological emergency, irrespective of how the emergency was initiated. According to para. 3.1 of GSR Part 7:

> "The goal of emergency preparedness is to ensure that an adequate capability is in place within the operating organization and at local, regional and national levels and, where appropriate, at the international level, for an effective response in a nuclear or radiological emergency".

This capability relates to an integrated set of infrastructural elements that include but are not limited to authority and responsibilities; organization and staffing; coordination; plans and procedures; tools, equipment and facilities; training, drills and exercises; and a management system. Maintaining adequate preparedness enables all the goals of emergency response (given in para. 3.2 of GSR Part 7 [14]) to be effectively met. The application of these requirements is also intended to mitigate the consequences of this emergency if such an emergency arises, despite all efforts made to prevent it. The requirements are intended to be applied by governments at various levels (national, regional, local), as well as by different response organizations, operating organizations and regulatory bodies. The requirements apply to all facilities, activities and sources with the potential for causing radiation exposure, environmental contamination or concern of the public warranting protective or other response actions.

2.3.10. Legal and regulatory framework

The main purpose of IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety [8], is to help Member States to develop laws and an effective regulatory body to adequately ensure the safety of nuclear facilities. The publication describes the essential aspects of the governmental and legal framework that enable adequate regulation at all stages in the lifetime of nuclear installations until their release from regulatory control and at any subsequent period of institutional control.

2.3.11. Transport safety

IAEA Safety Standards Series No. SSR-6 (Rev. 1), Regulations for the Safe Transport of Radioactive Material [23], establishes safety requirements for the transport of radioactive material by all modes of transport. The basic safety philosophy of these regulations is that under both normal and accident conditions and regardless of the mode of transport, safety is primarily assured by the packaging. During the transport of radioactive material, therefore, it is the package that serves to protect people, property and the environment from the harmful effects of ionizing radiation. The IAEA transport regulations are implemented worldwide through regional agreements, national legislations or both for surface transport (including road, rail and inland waterway) and through the International Maritime Organization (IMO) through the International Maritime Dangerous Goods (IMDG) Code [24] for sea transport.

2.4. SAFETY GUIDES CONSIDERED

Safety Guides provide recommendations on how to comply with the safety requirements. In this subsection, they are grouped under the following headings: siting, design, construction, commissioning and operation, nuclear fuel cycle facilities, radioactive waste management and decommissioning, leadership and management for safety, safety assessment of nuclear installations, emergency preparedness and response, legal and regulatory framework, and transport safety.

2.4.1. Siting of nuclear installations

The IAEA has developed several Safety Guides with recommendations to Member States on how to evaluate a proposed site for an NPP (and in general a nuclear installation) and, in particular, on how to characterize the site specific conditions pertinent to the safety of the NPP. First, IAEA Safety Standards Series No. SSG-35, Site Survey and Site Selection for Nuclear Installations [25], provides recommendations and guidance on establishing a systematic process for site survey and site selection from one of several preferred candidate sites. The necessary data at the different stages of the process are presented. Additionally, the issue of the unsuitability of a site is discussed. Other Safety Guides cover specific issues to be evaluated when aiming to site nuclear facilities:

(a) IAEA Safety Standards Series No. SSG-9 (Rev. 1), Seismic Hazards in Site Evaluation for Nuclear Installations [26], states:

  "The objective of this Safety Guide is to provide recommendations on how to meet the requirements established in SSR-1…in relation to the evaluation of hazards generated by earthquakes that might affect a nuclear installation site and, in particular, on how to determine the following:

  (a) The vibratory ground motion[2] hazards necessary to establish the design basis ground motions and other relevant parameters for the design and safety assessment of both new and existing nuclear installations;
  (b) The potential for, and the rate of, fault displacement phenomena that could affect the feasibility of a site for a new nuclear installation or the safe operation of an existing installation at a site;
  (c) The earthquake parameters necessary for assessing the associated geological and geotechnical hazards (e.g. soil liquefaction, landslides, differential settlements, collapse due to cavities and subsidence phenomena) and concomitant events (e.g. external flooding phenomena such as tsunamis and fires).

[2] In this Safety Guide, the terms 'vibratory ground motion' and 'ground motion' are synonymous. In some States, vibratory ground motion is called 'earthquake ground motion' or 'seismic ground motion'."

(b) IAEA Safety Standards Series No. SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations [27], provides "recommendations and guidance on how to comply with the safety requirements on assessing…hazards associated with meteorological and hydrological phenomena". It also "provides recommendations on how to determine the corresponding design bases for these natural hazards and it recommends measures for protection of the site against hazards of this type".
(c) IAEA Safety Standards Series No. SSG-21, Volcanic Hazards in Site Evaluation for Nuclear Installations [28], states:

"The objective of this Safety Guide is to provide recommendations and guidance on the assessment of volcanic hazards at a nuclear installation site, so as to enable the identification and comprehensive characterization of all potentially hazardous phenomena that may be associated with future volcanic events. These volcanic phenomena may affect the suitability of the selected site and some of them may determine corresponding design basis parameters for the installation."

(d) IAEA Safety Standards Series No. NS-G-3.6, Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants [29], provides "guidance on dealing with geotechnical engineering aspects that are important for the safety of nuclear power plants".
(e) IAEA Safety Standards Series No. SSG-79, Hazards Associated with Human Induced External Events in Site Evaluation for Nuclear Installations [30] states:

"The objective of this Safety Guide is to provide recommendations on evaluation of hazards associated with HIEEs [human induced external events] that could affect the safety of nuclear installations, in order to meet the requirements established in SSR-1…, in particular Requirements 6–9, 14 and 24. These hazards need to be considered in the selection and evaluation of sites for nuclear installations, in the design of new nuclear installations and in the operation of existing nuclear installations."

(f) IAEA Safety Standards Series No. NS-G-3.2, Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants [31], provides "guidance on the studies and investigations necessary for assessing the impact of a nuclear power plant on humans and the environment. It also provides guidance on the feasibility of an effective emergency response plan, in consideration of all the relevant site features.” In addition, it offers “guidance on investigations relating to population distribution, and on the dispersion of effluents in air, surface water and groundwater”.

2.4.2. Design

The IAEA has developed several Safety Guides with recommendations to Member States on how to comply with the design safety requirements. Examples include the following:

(a) IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants [32], concerning the identification of structures, systems, and components (SSCs) important to safety and their classification on the basis of their function and safety significance.
(b) IAEA Safety Standards Series No. SSG-52, Design of the Reactor Core for Nuclear Power Plants [33], “provides recommendations on the design of the reactor core to meet the requirements established in…SSR-2/1 (Rev. 1)”.
(c) IAEA Safety Standards Series No. SSG-56, Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants [34], “provides recommendations on how to meet the requirements of…SSR-2/1 (Rev. 1)… in relation to the cooling systems for nuclear power plants”.
(d) IAEA Safety Standards Series No. SSG-53, Design of the Reactor Containment and Associated Systems for Nuclear Power Plants [35], “provides recommendations on how to meet the requirements of…SSR-2/1 (Rev. 1)…in relation to the containment structures and systems for nuclear power plants”.
(e) IAEA Safety Standards Series No. SSG-34, Design of Electrical Power Systems for Nuclear Power Plants [36], “provides recommendations on the necessary characteristics of electrical power systems for nuclear power plants and of the processes for developing these systems, in order to meet the safety requirements of SSR-2/1 (Rev. 1)”.
(f) IAEA Safety Standards Series No. SSG-39, Design of Instrumentation and Control Systems for Nuclear Power Plants [37], “provides recommendations on the design of instrumentation and control (I&C) systems to meet the requirements established in...SSR-2/1 (Rev. 1)”.

(g) IAEA Safety Standards Series No. SSG-51, Human Factors Engineering in the Design of Nuclear Power Plants [38], states:

"This Safety Guide provides recommendations on the application of human factors engineering (HFE)[1] to meet the requirements established in… SSR-2/1 (Rev. 1).

[1] 'Human factors engineering' is engineering in which factors that could influence human performance and that could affect safety are understood and are taken into account, especially in the design and operation of facilities."

(h) IAEA Safety Standards Series No. SSG-62, Design of Auxiliary Systems and Supporting Systems for Nuclear Power Plants [39], "provides recommendations on how to meet the requirements established in… SSR-2/1 (Rev. 1)…in relation to the design of auxiliary systems and supporting systems for nuclear power plants".
(i) IAEA Safety Standards Series No. SSG-63, Design of Fuel Handling and Storage Systems for Nuclear Power Plants [40], provides "recommendations on how to meet the requirements established in...SSR-2/1 (Rev. 1)…in relation to the design of fuel handling and storage systems for nuclear power plants". With respect to protection against hazards in the design of NPPs, the following Safety Guides apply:
(j) IAEA Safety Standards Series No. SSG-64, Protection Against Internal Hazards in the Design of Nuclear Power Plants [41], "provides recommendations on how to meet the requirements established in... SSR-2/1 (Rev. 1)…in relation to protection against internal hazards in the design of land based stationary nuclear power plants with water cooled reactors".
(k) IAEA Safety Standards Series No. SSG-68, External Events Excluding Earthquakes in the Design of Nuclear Power Plants [42], provides guidance on assessing the impacts of a wide range of external events and the means of providing protection against them, including advice on the layout of the installation.
(l) IAEA Safety Standards Series No. SSG-67, Seismic Design and Qualification for Nuclear Power Plant [43], includes the general requirements of a seismic design, the required input, seismic design rules for SSCs, guidelines for seismic analysis and seismic qualification, minimum requirements for seismic instrumentation, and the seismic margin to be achieved by the design.
(m) IAEA Safety Standards Series No. SSG-69, Equipment Qualification for Nuclear Installations [44], applies to electrical, I&C, active mechanical

equipment and components associated with this equipment (e.g. seals, gaskets, lubricants, cables, connections, mounting/anchoring structures).

(n) IAEA Safety Standards Series No. NS-G-1.13, Radiation Protection Aspects of Design for Nuclear Power Plants [45], "addresses the provisions that should be made in the design of nuclear power plants in order to protect site personnel, the public and the environment against radiological hazards for operational states, decommissioning and accident conditions". Accident conditions are discussed for the protection of site personnel. This Safety Guide is currently under revision.

2.4.3. Construction of nuclear installations

IAEA Safety Standards Series No. SSG-38, Construction for Nuclear Installations [46], provides recommendations and guidance based on international good practices in the construction of NPPs (and other nuclear installations), "as currently followed in Member States, that will enable construction to be of high quality, consistent with the design requirements, and as agreed by the regulatory body in issuing the authorization for construction".

2.4.4. Commissioning and operation

The IAEA has developed several Safety Guides with recommendations to Member States on how to comply with the safety requirements established for commissioning and operation of NPPs, as briefly described next.

(a) IAEA Safety Standards Series No. SSG-28, Commissioning for Nuclear Power Plants [47]. This safety standard makes "recommendations on the basis of international good practices in commissioning for nuclear power plants, as currently followed in Member States".

(b) IAEA Safety Standards Series No. SSG-76, Conduct of Operations at Nuclear Power Plants [48]. This safety standard is concerned with:

> "(a) The structure of the operations department of a nuclear power plant;
> (b) Setting high standards of performance and making safety related decisions in an effective manner;
> (c) Conducting control room activities in a thorough and professional manner;
> (d) Maintaining a nuclear power plant within the established operational limits and conditions."

(c) IAEA Safety Standards Series No. SSG-72, The Operating Organization for Nuclear Power Plants [49], states that it “provides specific recommendations on establishing and maintaining the operating organization of nuclear power plants to ensure effective management for safety”.
(d) IAEA Safety Standards Series No. SSG-70, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants [50]. This standard makes recommendations for the development and implementation of operational limits and conditions (OLCs) and operating procedures, a key concern being that they should be consistent with each other.
(e) IAEA Safety Standards Series No. SSG-71, Modifications to Nuclear Power Plants [51]. This standard relates to the control of plant modifications at an NPP so that the configuration of the plant always remains under control and the modified configuration conforms to the NPP’s operating licence.
(f) IAEA Safety Standards Series No. SSG-73, Core Management and Fuel Handling for Nuclear Power Plants [52], is complemented by SSG-27 (Rev. 1), Criticality Safety in the Handling of Fissile Material [53], which deals with (among other things) the handling of fresh and spent nuclear fuel (SNF).
(g) IAEA Safety Standards Series No. SSG-74, Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants [54]. The standard is concerned with the measures, including organizational and administrative arrangements, that are needed for SSCs important to safety to perform as intended.
(h) IAEA Safety Standards Series No. SSG-75, Recruitment, Qualification and Training of Personnel for Nuclear Power Plants [55]. This standard deals specifically with those aspects of qualification and training that are important to the safe operation of nuclear power plants. Some parts or all of this Safety Guide may also be used, with due adaptation, as a guide to the recruitment, selection, training and qualification of staff for other nuclear installations (such as research reactors or nuclear fuel cycle facilities).
(i) IAEA Safety Standards Series No. SSG-13, Chemistry Programme for Water Cooled Nuclear Power Plants [56], provides recommendations on the control of coolant chemistry so that “structures, systems and components important to safety are available to perform their function in accordance with the assumptions and the intent of the design.”
(j) IAEA Safety Standards Series No. SSG-48, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants [57]. This standard “supplements and provides recommendations on meeting the requirements relating to ageing management and long term operation that are established in SSR-2/1 (Rev. 1)…and SSR-2/2 (Rev. 1)”.

(k) IAEA Safety Standards Series No. SSG-50, Operating Experience Feedback for Nuclear Installations [58]. This standard aims to prevent or minimize accident risk by learning from events that have already occurred at the installation or elsewhere.
(l) IAEA Safety Standards Series No. SSG-77, Protection Against Internal and External Hazards in the Operation of Nuclear Power Plants [59]. This standard "provides new or updated recommendations derived from enhanced understanding of operational aspects of hazards and combinations of hazards."
(m) IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants [60]. The standard "provides recommendations on the design of the reactor core to meet the requirements established in…SSR-2/1 (Rev. 1)"

2.4.5. Nuclear fuel cycle facilities

The IAEA has issued the following Safety Guides applicable to nuclear fuel cycle facilities:

(a) IAEA Safety Standards Series No. SSG-27 (Rev. 1), Criticality Safety in the Handling of Fissile Material [53]. In addition to being relevant to fuel handling at NPPs, this Safety Guide also covers other nuclear fuel cycle facilities as follows:

"The recommendations provided in this Safety Guide cover criticality safety during operational states and during conditions referred to as 'credible abnormal conditions', from initial design, through commissioning and operation, to decommissioning. It also applies to the design, operation and post closure stages of waste disposal facilities. This Safety Guide also provides recommendations on planning the emergency response to a criticality accident".

(b) IAEA Safety Standards Series No. SSG-5, Safety of Conversion Facilities and Uranium Enrichment Facilities [61], addresses the safety recommendations for conversion, purification and enrichment (low enriched uranium with up to 6% ^{235}U concentration) facilities.
(c) IAEA Safety Standards Series No. SSG-6, Safety of Uranium Fuel Fabrication Facilities [62], covers oxide fuel (ceramic fuel) fabrication, but excludes metallic and other fuels.
(d) IAEA Safety Standards Series No. SSG-7, Safety of Uranium and Plutonium Mixed Oxide Fuel Fabrication Facilities [63], addresses the safety of MOX

(uranium and plutonium mixed oxide) fuel fabrication facilities. It does not deal with any impact that the manufactured fuel assemblies may have on safety for the reactors in which they are to be used.

(e) IAEA Safety Standards Series No. SSG-42, Safety of Nuclear Fuel Reprocessing Facilities [64], provides recommendations on meeting requirements for spent fuel reprocessing facilities using the plutonium uranium extraction reduction (PUREX) process to reprocess fuels containing uranium and plutonium on a commercial scale. SSG-42 does not specifically address thorium breeder reprocessing (THOREX) and partitioning of radionuclides other than uranium and plutonium (such as SANEX, GANEX and UNEX processes).

(f) IAEA Safety Standards Series No. SSG-43, Safety of Nuclear Fuel Cycle Research and Development Facilities [65], provides guidance for different research and development (R&D) facilities related to the nuclear fuel cycle. The guidance, however, is not applicable to industrial or pilot plants.

2.4.6. Radiation protection and safety

2.4.6.1. Protection of workers

IAEA Safety Standards Series No. GSG-7, Occupational Radiation Protection [66], provides guidance on fulfilling the requirements in GSR Part 3 [10] for the protection of workers in planned, emergency and existing exposure situations. It provides general guidance on the development of occupational radiation protection programmes, as appropriate, for the sources of radiation likely to be encountered in the workplace to fulfil the management's responsibility for protection and safety. Detailed guidance is also provided on the monitoring and assessment of workers' exposure due to external radiation sources and from intakes of radionuclides.

The IAEA is preparing a Safety Guide on the protection of workers against exposure due to radon, which will provide guidance on the management and recording of exposure due to radon in workplaces where other sources of radiation are present.

2.4.6.2. Protection of the public and the environment

There are four Safety Guides that provide generic guidance on meeting the requirements established in GSR Part 3 [10] for the protection of the public and the environment.

(a) IAEA Safety Standards Series No. GSG-8, Radiation Protection of the Public and the Environment [67], provides guidance on the implementation of the requirements established in GSR Part 3 in relation to protection of the public and the environment against radiation risks. It provides generic guidance on the application of the radiation protection principles of justification, optimization and dose limitation. The publication covers protection of the public and the environment in all exposure situations — planned, emergency and existing.
(b) IAEA Safety Standards Series No. GSG-9, Regulatory Control of Radioactive Discharges to the Environment [68], provides recommendations on the application of the safety requirements established in GSR Part 3 to the regulatory control of discharges. It provides a structured approach to controlling radiation exposures of the public resulting from discharges from normal operation of facilities and activities for the optimization of protection and safety. Guidance is also provided on authorization for discharges, demonstrating compliance with the authorization and enforcing the authorization.
(c) IAEA Safety Standards Series No. GSG-10, Prospective Radiological Environmental Impact Assessment for Facilities and Activities [69], provides recommendations and guidance on a general framework for performing prospective radiological impact assessments for facilities and activities, to estimate and control the radiological effects on the public and the environment. This radiological impact assessment is intended for planned exposure situations as part of the authorization process and, when applicable, as part of a government decision making process for facilities and activities. It covers situations for both exposures expected to occur during normal operation and potential exposures.
(d) IAEA Safety Standards Series No. RS-G-1.7 (under revision), Application of the Concepts of Exclusion, Exemption and Clearance [70], addresses safety in relation to potential release of material for unconditional reuse or for non-radiological disposal during the normal operation of a facility, during decommissioning or as part of remediation activities following a nuclear or radiological emergency; it may also be applicable to other situations. RS-G-1.7 is currently being revised, and the revision will specifically focus on the application of the concept of clearance.

2.4.7. Radioactive waste management and decommissioning

The Safety Requirements publications on radioactive waste management (GSR Part 5 and SSR-5 [12, 21]) and on decommissioning (GSR Part 6 [13]) are supported by several Safety Guides. The guides of greatest relevance to EIDs and waste deriving from EIDs are the following:

(a) IAEA Safety Standards Series No. GSG-1, Classification of Radioactive Waste [71]. This standard "provides guidance on the classification of the whole range of radioactive waste: from spent nuclear fuel, when it is considered radioactive waste, to waste having such low levels of activity concentration that it is not required to be managed or regulated as radioactive waste."
(b) IAEA Safety Standards Series No. GSG-16, Leadership, Management and Culture for Safety in Radioactive Waste Management [72].
(c) IAEA Safety Standards Series No. SSG-40, Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors [73]. It provides "recommendations on how to meet the requirements for the predisposal management of radioactive waste generated at nuclear power plants".
(d) IAEA Safety Standards Series No. SSG-41, Predisposal Management of Radioactive Waste from Nuclear Fuel Cycle Facilities [74]. It "provides guidance on the predisposal management of all types of radioactive waste (including spent nuclear fuel declared as waste and high level waste) generated at nuclear fuel cycle facilities."
(e) IAEA Safety Standards Series No. SSG-45, Predisposal Management of Radioactive Waste from the Use of Radioactive Material in Medicine, Industry, Agriculture, Research and Education [75]. It "focuses on waste generated at small to moderate sized facilities, such as hospitals and research centres, where radioactive waste is not usually generated in bulk quantities."
(f) IAEA Safety Standards Series No. WS-G-6.1, Storage of Radioactive Waste [76]. It covers "the storage of solid, liquid and gaseous radioactive wastes in a wide range of facilities, including those at which waste is generated, treated and conditioned."
(g) IAEA Safety Standards Series No. GSG-3, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste [77]. It provides "recommendations and guidance on the development and review of the safety case and supporting safety assessment prepared or conducted for a predisposal waste management facility or activity.
(h) IAEA Safety Standards Series No. SSG-15 (Rev. 1), Storage of Spent Nuclear Fuel [78]. It covers all types of storage facility and all types of spent fuel from nuclear power plants and research reactors. It takes into consideration the longer storage periods that have become necessary owing to delays in the development of disposal facilities and the decrease in reprocessing activities.

(i) IAEA Safety Standards Series No. SSG-1, Borehole Disposal Facilities for Radioactive Waste [79], provides guidance on meeting and demonstrating compliance with GSR Part 5 in the development of borehole disposal facilities.
(j) IAEA Safety Standards Series No. SSG-31, Monitoring and Surveillance of Radioactive Waste Disposal Facilities [80], provides guidance for the monitoring and surveillance of radioactive waste disposal facilities throughout their lifetime. It addresses the different aims of monitoring and surveillance at the various stages of the lifetime of a disposal facility, from initiation of work on a candidate site to the period after closure of the disposal facility.
(k) IAEA Safety Standards Series No. SSG-14, Geological Disposal Facilities for Radioactive Waste [81], provides guidance on meeting and demonstrating compliance with GSR Part 5 in the development of geological disposal facilities.
(l) IAEA Safety Standards Series No. SSG-23, The Safety Case and Safety Assessment for the Disposal of Radioactive Waste [82]. It elaborates on "the safety requirements in respect of the safety case and supporting safety assessment for the disposal of radioactive waste."
(m) IAEA Safety Standards Series No. SSG-29, Near Surface Disposal Facilities for Radioactive Waste [83], provides guidance on meeting and demonstrating compliance with GSR Part 5 in the development of near-surface disposal facilities.
(n) IAEA Safety Standards Series No. SSG-47, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities [84] states:

    "This Safety Guide provides guidance on meeting the safety requirements applicable to decommissioning, primarily those established in GSR Part 6…but also in other Safety Requirements publications. This Safety Guide addresses decommissioning considerations and actions for the safe decommissioning of nuclear power plants, research reactors and other nuclear fuel cycle facilities."

(o) IAEA Safety Standards Series No. WS-G-5.1, Release of Sites from Regulatory Control on Termination of Practices [85]. It provides advice on "the release of sites or parts of sites from regulatory control after a practice has been terminated."
(p) IAEA Safety Standards Series No. WS-G-5.2, Safety Assessment for the Decommissioning of Facilities Using Radioactive Material [86]. This Safety Guide provides recommendations for the development and review

of safety assessments in support of a decommissioning plan and for the individual decommissioning activities within this plan. Emphasis is placed on the importance of the graded approach.

2.4.8. Leadership and management for safety

The safety requirements pertaining to leadership and management for safety (GSR Part 2 and SSR-2/2 (Rev. 1) [9, 19]) are complemented by the following Safety Guides:

(a) IAEA Safety Standards Series No. GS-G-3.1, Application of Management System for Facilities and Activities [87]. This Safety Guide provides generic guidance for establishing, implementing, assessing and continually improving a management system that integrates safety, health, environmental, security, quality and economic elements to meet the safety requirements established in GSR Part 2. The guide is applicable throughout the lifetime of facilities and for the entire duration of activities in normal, transient and emergency situations.
(b) IAEA Safety Standards Series No. GS-G-3.5, The Management System for Nuclear Installations [88]. This Safety Guide provides recommendations and guidance supplementary to those provided in GS-G-3.1 for establishing, implementing, assessing and continually improving a management system that integrates elements of safety, health, environment, security, quality and economics. All of the topics covered correspond to requirements established in GSR Part 2 [9]. The guide is applicable throughout the lifetime of a nuclear installation.

2.4.9. Safety assessment of nuclear installations

The IAEA developed the following Safety Guides with recommendations to Member States on how to comply with the safety assessment requirements for NPPs:

(a) IAEA Safety Standards Series No. SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [89]. Its objective is "to provide recommendations and guidance for designers, operating organizations, regulatory bodies and technical support organizations on performing deterministic safety analysis and on its application to nuclear power plants".
(b) IAEA Safety Standards Series No. SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [90]. Its objective "is to provide recommendations for meeting the requirements…

in performing or managing a Level 1 PSA [probabilistic safety assessment] project for a nuclear power plant and using it to support its safe design and operation".

(c) IAEA Safety Standards Series No. SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [91]. This Safety Guide provides recommendations for meeting the GSR Part 4 requirements "in performing or managing a Level 2 PSA [probabilistic safety assessment] project for a nuclear power plant" and complements SSG-3 [90].
(d) IAEA Safety Standards Series No. SSG-25, Periodic Safety Review for Nuclear Power Plants [92]. It provides recommendations and guidance on the conduct of a periodic safety review (PSR) of an existing nuclear power plant…PSR is a comprehensive safety review of all important aspects of safety, carried out at regular intervals, typically every ten years".
(e) IAEA Safety Standards Series No. NS-G-2.13 (under revision), Evaluation of Seismic Safety for Existing Nuclear Installations [93]. It "provides recommendations in relation to the seismic safety evaluation of existing nuclear installations".

2.4.10. Emergency preparedness and response

The IAEA developed five Safety Guides with recommendations to Member States on fulfilling the requirements established in GSR Part 7 [14]. They are as follows:

(a) IAEA Safety Standards Series No. GS-G-2.1 (under revision), Arrangements for Preparedness for a Nuclear or Radiological Emergency [94]. Its objectives are, inter alia, to respond to Member States' requests for guidance on meeting certain aspects of the safety requirements for emergency preparedness and response and to describe appropriate responses to a range of emergencies. The guidance includes hazard assessment[5], criteria for determining the emergency preparedness category (EPC)[6] of facilities and

[5] Referred to as 'threat assessment' in GS-G-2.1 [94].
[6] Referred to as 'threat category' in GS-G-2.1 [94].

activities, and suggested sizes of emergency planning zones for facilities in emergency preparedness categories I and II[7].

(b) IAEA Safety Standards Series No. GSG-2 (under revision), Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency [95]. Its objectives are: (i) to present a coherent set of generic criteria (expressed numerically in terms of radiation dose) that form a basis for developing the operational levels needed for decision making concerning protective actions, and other response actions necessary to meet the emergency response objectives; and (ii) to propose a basis for a plain language explanation of the criteria for the public and for public officials that addresses the risks to human health of radiation exposure.

(c) IAEA Safety Standards Series No. GSG-11, Arrangements for the Termination of a Nuclear or Radiological Emergency [96]. It provides guidance and recommendations on arrangements to be made at the preparedness stage, as part of overall emergency preparedness, for the termination of a nuclear or radiological emergency and the subsequent transition from the emergency exposure situation to either a planned exposure situation or an existing exposure situation. It elaborates the prerequisites that need to be fulfilled so that responsible authorities can declare the nuclear or radiological emergency ended, and it gives detailed guidance on adapting and lifting protective actions.

(d) IAEA Safety Standards Series No. GSG-14, Arrangements for Public Communication in Preparedness and Response for a Nuclear or Radiological Emergency [97]. Its objectives are to provide recommendations on meeting requirements relating to arrangements for public communication in preparedness and response for a nuclear or radiological emergency. In addition, this Safety Guide provides recommendations to Member States on arrangements to be made at the preparedness stage for communication with the public and the news media for the purpose of mitigating the adverse consequences of such an emergency on human life, health, property and the environment, as well as on activating these arrangements in an emergency response. It further provides recommendations on the coordination of response organizations and other authorities that provide official information on preparedness and response for such emergency.

---

[7] GS-G-2.1 [94] states:

"The Requirements...and the guidance in this publication are often specified for the threat categories [I to V]. Threat categories I, II and III represent decreasing levels of threat at major facilities and therefore correspond to decreasing stringency of requirements for emergency preparedness and response. Facilities in threat categories I and II warrant extensive on-site and off-site arrangements for emergency preparedness."

(e) IAEA Safety Standards Series No. SSG-65, Preparedness and Response for a Nuclear or Radiological Emergency Involving the Transport of Radioactive Material [98]. It provides "recommendations on emergency preparedness and response for the transport of radioactive material. These recommendations form the basis of achieving the goals of emergency response described in GSR Part 7."

2.4.11. Legal and regulatory framework

The IAEA has issued several Safety Guides with recommendations to Member States on how to develop the governmental and legal framework for establishing a regulatory body and for taking other actions necessary for effective regulatory control, as follows:

(a) IAEA Safety Standards Series No. GSG-12, Organization, Management and Staffing of the Regulatory Body for Safety [16]. It outlines recommendations on the organizational structure, management and staffing of a regulatory body to support this body in carrying out its responsibilities and functions efficiently, effectively and in an independent manner.
(b) IAEA Safety Standards Series No. GSG-13, Functions and Processes of the Regulatory Body for Safety [17]. It outlines recommendations on the regulatory body's core functions and the associated processes to implement these functions.
(c) IAEA Safety Standards Series No. SSG-12, Licensing Process for Nuclear Installations [99]. It outlines recommendations on how the licensing process should be applied at the various stages of the lifetime of a nuclear installation, with discussion of the topics and required documents to be considered at each stage.
(d) IAEA Safety Standards Series No. GSG-6, Communication and Consultation with Interested Parties by the Regulatory Body [15]. It provides recommendations on meeting the safety requirements concerning communication and consultation with the public and other interested parties by the regulatory body. It addresses communication and consultation about the possible radiation risks associated with facilities and activities, and about processes and decisions of the regulatory body.
(e) IAEA Safety Standards Series No. SSG-16 (Rev. 1), Establishing the Safety Infrastructure for a Nuclear Power Programme [100]. It provides guidance for the establishment of the safety infrastructure for a nuclear power programme in the first three phases of such establishment: (1) safety infrastructure considerations before a decision to launch a nuclear power programme; (2) safety infrastructure preparatory work for the construction

of an NPP; and (3) safety infrastructure activities to implement the first NPP. It provides a roadmap in the form of 197 actions for the progressive application of the relevant IAEA safety standards when developing the safety infrastructure.

2.4.12. Transport safety

The implementation of the IAEA Transport Regulations is supported by seven IAEA Safety Guides. While SSR-6 (Rev. 1) [23] specifies what is required and has to be met during transport, the related Safety Guides describe how these requirements can be fulfilled in practice, as follows:

(a) IAEA Safety Standards Series No. SSG-26 (Rev. 1), Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material [101]. It provides specific guidance to each of the requirements of SSR-6 (Rev. 1) [23], where appropriate.
(b) IAEA Safety Standards Series No. SSG-65, Preparedness and Response for a Nuclear or Radiological Emergency Involving the Transport of Radioactive Material [98]. It provides "recommendations on emergency preparedness and response for the transport of radioactive material. These recommendations form the basis of achieving the goals of emergency response described in GSR Part 7". It offers guidance on various aspects of emergency planning and preparedness for dealing effectively and safely with transport accidents involving radioactive material, including the assignment of responsibilities. It reflects the requirements specified in SSR-6 (Rev. 1) [23] and GSR Part 3 [10].
(c) IAEA Safety Standards Series No. TS-G-1.5, Compliance Assurance for the Safe Transport of Radioactive Material [102]. It is intended "to assist competent authorities in the development and maintenance of compliance assurance programmes in connection with the transport of radioactive material, and to assist applicants, licensees and organizations in their interactions with competent authorities."
(d) IAEA Safety Standards Series No. TS-G-1.4, The Management System for the Safe Transport of Radioactive Material [103]. It provides guidance on implementing the requirements in GSR Part 3 [10] "for establishing, implementing, assessing and continually improving a management system for the transport of radioactive material" and provides guidance on implementing the requirements established in IAEA Safety Standards No. TS-R-1, Regulations for the Safe Transport of Radioactive Material (superseded by SSR-6 (Rev. 1) [23]) "on quality assurance and quality assurance programmes within the management system for transport" [103].

It applies to “management systems for all activities relating to the transport of radioactive material, including, but not limited to, the design, fabrication, assembly, inspection, test, maintenance…and disposal of radioactive material packagings” [103].

(e) IAEA Safety Standards Series No. TS-G-1.3, Radiation Protection Programmes for the Transport of Radioactive Material [104]. It provides “guidance on meeting the requirements for the establishment of RPPs [radiation protection programmes] for the transport of radioactive material, to optimize radiation protection in order to meet the requirements for radiation protection that underlie the Transport Regulations”.
(f) IAEA Safety Standards Series No. SSG-33 (Rev. 1), Schedules of Provisions of the IAEA Regulations for the Safe Transport of Radioactive Material [105]. It aims “to provide information to aid users in determining the correct package type and the appropriate operational and administrative requirements to be applied.”
(g) IAEA Safety Standards Series No. SSG-66, Format and Content of the Package Design Safety Report for the Transport of Radioactive Material [106]. Its objective is “to provide recommendations on the preparation of a PDSR [package design safety report] to demonstrate compliance of a package design for the transport of radioactive material with the Transport Regulations.”

The applicability review for transport safety presented in Section 4 has been limited to the consideration of SSR-6 (Rev. 1) [23]. SSG-65 [98] was considered under emergency preparedness and response. However, the gaps identified in Section 4 also affect the applicability of the supporting Safety Guides because of the overarching nature of the gaps.

Other requirements that may result from specific national regulations for surface transport or from other non-class 7 specific IMO regulations for sea transport, have not been considered as part of this applicability review.

2.5. SAFETY, SECURITY AND SAFEGUARDS INTERFACES

2.5.1. The 3S concept

Requirement 8 (interfaces of safety with security and safeguards) of SSR-2/1 (Rev. 1) [3] states:

“Safety measures, nuclear security measures and arrangements for the State system of accounting for, and control of, nuclear material for a nuclear power plant shall be designed and implemented in an integrated manner so that they do not compromise one another."

This integrated approach to safety, security and safeguards is known as the 3S concept. Two IAEA publications provide guidance in this respect:

(a) From a nuclear security perspective, IAEA Nuclear Security Series No. 13, Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5) [107], recommends that physical protection and its interfaces with safety and nuclear material accountancy and control should be taken into account "as early as possible".
(b) From a nuclear safeguards perspective, Ref. [108] para. 87 states that inspectors' activities need to be carried out in a manner designed to avoid affecting the safety of the nuclear facility.

Thus, the 3S concept needs to be implemented through proper consideration of the interfaces between safety, security and safeguards.

Sections 2.5.2 and 2.5.3 provide a summary of the key elements of the IAEA safeguards and security frameworks that need to be considered in the definition of the interface with safety. The relevant aspects of these frameworks are described in many IAEA publications and are briefly explained in this publication to provide the basis of the assessment described in Section 5. A more detailed examination of the 3S interfaces is presented in Section 5.

2.5.2. Brief description of international safeguards framework

The objective of IAEA safeguards is to deter the spread of nuclear weapons by the early detection of the misuse of nuclear material or technology. This provides credible assurances that Member States are honouring their legal obligations to restrict the use of nuclear material and technology to peaceful purposes. IAEA safeguards are a set of technical measures applied by the IAEA with the aim of independently verifying that nuclear technologies are not misused and that nuclear material is not diverted for non-peaceful uses. Member States accept these measures through the conclusion of safeguards agreements with the IAEA.

The implementation of IAEA safeguards follows an annual cycle and comprises four main processes, which are described in Sections 2.5.2.1–2.5.2.4.

2.5.2.1. Collection and evaluation of safeguards relevant information

The IAEA collects, processes and reviews all available safeguards relevant information about a State to evaluate its consistency with the State's declarations about its nuclear fuel cycle programme and relevant activities.

2.5.2.2. Development of a safeguards approach for a State

A safeguards approach for a State includes the safeguards measures used to achieve the technical objectives for verifying the State's declarations on their nuclear fuel cycle programme and relevant activities.

2.5.2.3. Planning, conducting and evaluating safeguards activities

The IAEA develops a plan specifying the safeguards activities to be conducted both in the field and at the IAEA's headquarters. Once an activity has taken place, the IAEA evaluates the extent to which it has reached the technical objectives and identifies any inconsistencies that might need to be followed up.

2.5.2.4. Drawing of safeguards conclusions

The safeguards conclusions drawn by the IAEA are based on its independent verification and findings. They are the final product of the annual safeguards implementation cycle and provide credible assurances to the international community that States are abiding by their safeguards obligations.

The IAEA safeguards activities are based on assessments of the correctness and completeness of a State's declared nuclear material, nuclear fuel cycle programme and relevant activities. The IAEA verification measures include on-site inspections, visits and ongoing monitoring and evaluation[8] and may be considered to comprise two types of activity:

(a) The verification of State reports of declared nuclear material and activities. These measures are mainly based on nuclear material accountancy, complemented by containment and surveillance techniques, such as tamper-proof seals and cameras that the IAEA installs at facilities.
(b) Improvement of the IAEA's knowledge of the State's overall nuclear fuel cycle through broader information and complementary access. This includes activities covered by the Additional Protocol [109], which complements all

[8] More detailed information is available at https://www.iaea.org/publications/factsheets/iaea-safeguards-overview

safeguards agreements. The measures enhance the IAEA's ability to not only verify the non-diversion of declared nuclear material but also provide credible assurances as to the absence of undeclared nuclear material and activities in a State. The Additional Protocol is a legal document granting the IAEA complementary access authority and additional safeguards relevant information to that provided in underlying safeguards agreements.

Safeguards by design (SBD) is the process of including safeguards considerations throughout the lifetime of a nuclear facility; from the initial conceptual design through construction and operation (including design modifications) and into decommissioning. SBD has two main objectives: (1) avoiding costly and time consuming retrofits or redesigns of new nuclear facilities to accommodate IAEA safeguards; and (2) making the implementation of IAEA safeguards more effective and efficient at such facilities, reducing the burden for the operator, the State and the IAEA. With the aims of mitigating the potential for negative impact on the facility licensing process and helping to build public confidence, SBD seeks to reduce the impact of IAEA safeguards on the design and construction cost and schedule. The SBD concept and practices are under the continuous focus of the IAEA. Current SBD practices and considerations for different types of facility are summarized in a dedicated series of publications[9] developed by the IAEA.

2.5.3. Brief description of the security framework

2.5.3.1. Introduction to nuclear security

Nuclear security focuses on "the prevention and detection of, and response to, theft, sabotage, unauthorized access and illegal transfer or other malicious acts involving nuclear material and other radioactive substances and their associated facilities" [110]. Nuclear security focuses on criminal or intentional unauthorized acts and other associated activities. The nuclear security programme of the IAEA has the goal to protect people, property and the environment against the malicious use of nuclear or other radioactive material, as well as sabotage of nuclear facilities, including NPPs. In general, every State needs to establish and implement an overall nuclear security regime to prevent nuclear and radioactive material from ever falling into the wrong hands and to protect nuclear facilities

[9] The SBD series can be found at https://www.iaea.org/topics/assistance-for-states/safeguards-by-design-guidance. It covers international safeguards in design for various nuclear facilities (nuclear reactors, fuel fabrication plants, long term spent fuel management facilities, uranium conversion plants, reprocessing plants, uranium enrichment plants).

and activities from sabotage [110]. The IAEA assists Member States to improve nuclear security and, thus, reduce both the threat and the consequences of malicious acts involving nuclear and other radioactive material.

The most important elements of nuclear security are physical protection and cyber protection. These security measures are the basis of robust defence of nuclear and/or radioactive materials and their associated facilities [110–113].

2.5.3.2. Legal and regulatory framework for nuclear security

The legal and regulatory framework in a State is an important part of the implementation of effective nuclear security. It includes international legal instruments (binding and non-binding) that form the international legal framework for nuclear security. The following is a brief list of such international legal instruments relevant for the protection of nuclear and other radioactive material and nuclear facilities, including EIDs:

— Convention on the Physical Protection of Nuclear Material (CPPNM) and Its 2005 Amendment [114, 115];
— Convention on Early Notification of a Nuclear Accident [116];
— Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency [117];
— United Nations (UN) Security Council Resolutions 1373 (2001) and 1540 (2004) [118, 119];
— International Convention for the Suppression of Acts of Nuclear Terrorism (ICSANT) [120];
— Code of Conduct on the Safety and Security of Radioactive Sources [121].

Commitments made by a number of States within the international Nuclear Security Summit process (during the meetings in Washington, DC, USA, in 2010 [122]; Seoul, Republic of Korea, in 2012 [123]; The Hague, Netherlands, in 2014 [124]; and Washington, DC, USA, in 2016 [125]) also serve as an important tool to enhance nuclear security at a global level. To meet the obligations of the international legal framework on nuclear security, a State needs to develop national laws and regulations and then establish a competent authority mandate to implement them. An effective nuclear security regime needs all States to recognize the importance of the international legal framework and its corresponding obligations, including States with active nuclear programmes and those conducting more limited nuclear activities. Any State, with or without nuclear and other radioactive materials or activities within the borders of its territory, may be used as a transit country for nuclear material.

2.5.3.3. *Key functions of physical protection*

The key functions of physical protection (deterrence, detection, delay, response) [114, 115] are the same in every nuclear facility. Nevertheless, for a nuclear facility or a process involving nuclear or radioactive material, the actual realization of these functions will be highly dependent on the design of the facility or process. Security by design aims to reduce security risks at the source through an approach that considers security systematically and consistently through all the phases in the lifetime of the facility or process. Another important factor is the location of the site and its complexity, which may result in the physical protection measures being equally complex. Implementation of strict nuclear material accounting rules will support the detection function against insider threats.

The key functions of physical protection are as follows:

(a) Deterrence

Deterrence is a function of a nuclear security physical protection system (PPS). As stated in IAEA Nuclear Security Series No. 27-G, Physical Protection of Nuclear Material and Nuclear Facilities (Implementation of INFCIRC/225/Revision 5) [111], deterrence can be achieved by making the target less attractive for potential adversaries (e.g. they decide not to attack because of the low probability of success). Observable protection measures might be used to promote the deterrence[10].

Implementation of a policy for determining the trustworthiness of all people working on a nuclear facility also serves as strong deterrence against the possible insider adversary. The effectiveness of deterrence needs to be evaluated (adequate exercises, comparisons, audits).

(b) Detection

If the perpetrator is not deterred by the PPS, the process of detection starts with the alarm being triggered by the recognition of a malicious act, unauthorized act or the presence of an adversary. The detection process is considered to be completed if the triggered alarm is assessed.

The main factors influencing the detection are the capabilities of the detection measures (e.g. sensors, alarm signal activation), alarm reporting

[10] These measures include, for instance, the presence of patrolling armed guards (visible), controlled access points, fences, proper lighting during the night and visible barriers (e.g. bars on windows, vehicle barriers).

and assessment, as well as the performance of the relevant staff having a role in detection. Hence, the technology can significantly increase the efficiency of the detection process and whenever used needs to provide an ability to sense and assess [111].

Measures such as a two-person rule or remote monitoring may be applied in the central alarm station to reduce the insider threat and may be needed for some central alarm station functions, such as putting a sensor into access mode (non-secure) or remotely opening a high security door [110–112].

(c) Delay

After detection has taken place, the subsequent integral function of a nuclear security regime is the ability of a PPS to obstruct by physical barriers, security forces and/or active delay systems. Therefore, according to IAEA Nuclear Security Series No. 27-G [111], delay is "the function of the physical protection system that seeks to slow an adversary's progress towards a target, thereby providing more time for effective response."

The main means to accomplish delay are distances and areas on the way of adversaries and barriers that need to be defeated or bypassed. Estimated delay times are important considerations for designing the PPS for NPPs (including EIDs). Guards or response forces may provide further delay if they are appropriately positioned, armed and protected.

In this context, it needs to be highlighted that the physical protection measures that are encountered by the perpetrator before a detection event do not provide additional time for the emergency and/or response force. For remote locations, where the response force may not be located close to the facility, delay measures may provide sufficient delay to enable the response force to reach the facility and prevent the unauthorized act.

(d) Response

Response involves the capability of the competent authorities (including regulatory bodies, law enforcement, customs and border control, intelligence and security agencies or health agencies), facility operators and other relevant stakeholders in the nuclear security regime. The first and second essential elements in IAEA Nuclear Security Series No. 20, Nuclear Security Fundamentals, Objective and Essential Elements of a State's Nuclear Security Regime [110], define the role of the State in establishing an effective and appropriate nuclear security regime and the competent authorities designated as part of the regime. This is to say, each State has the responsibility to define the objectives and tasks of each of the stakeholders involved in the nuclear security response [111] to respond to malicious acts related to nuclear security or threats (unauthorized removal of materials or sabotage) in an effective and coordinated manner. As described in IAEA Nuclear Security Series No. 27-G [111]:

> "Response is the function of the physical protection system that seeks to interrupt and neutralize an adversary before the completion of a malicious act. Guards are assigned responsibility for controlling access, escorting individuals, monitoring and assessing alarms in the central alarm station, patrolling and/or providing the initial response on detection of a potential adversary. These guards may or may not be prepared or permitted to provide an armed response. The response force consists of persons on-site or off-site who are armed and appropriately equipped and trained to interrupt and neutralize an adversary attempting unauthorized removal or an act of sabotage."

The response, as the other functions mentioned above, is to be implemented with a graded approach, which implies that the application of proportionate physical protection measures based on the potential consequences of a malicious act is required. This principle is applicable not only to the response domain but also to physical protection measures in general. The application of physical protection measures is proportional to the potential consequences of a malicious act[11].

3. IDENTIFICATION OF AREAS OF NOVELTY FOR EIDs

This section presents areas of novelty at the various stages in the lifetime of an EID. For each stage, the areas of novelty are presented in the following order:

(1) Group 1: General areas of novelty. This category captures areas of novelty of EIDs, regardless of size, technology, and use of multiple modules and modularity.
(2) Group 2: Areas of novelty related to small size, multiple modules or modularity. They are independent of the technology, that is, they are

[11] It implies consideration of various factors, such as type of material, isotopic composition, chemical form, radiation level and quantity of material.

common to water cooled and non-water cooled EIDs. This category typically addresses areas of novelty that are common to SMRs.

(3) Group 3: Areas of novelty specific to non-WCRs. They are specific to this type of reactor, regardless of the size of the NPP (large or SMR).

(4) Group 4: Preliminary areas of novelty that are specific to TNPPs. These may be SMRs or larger NPPs, although, by their very nature, they will naturally be size limited and possibly modular. This is a field that was not fully explored by this review, so initial insights about areas of novelty are presented.

There are also some areas of novelty that are specific to new designs of WCRs. Since this type of reactor is associated with SMRs, they are included in Group 2. The four groups are associated with EIDs, but they are not necessarily mutually exclusive. For example, the areas that are typically related to SMRs, such as small size and modularity, are presented in Group 2, and those specifically associated with non-WCRs in Group 4. These four groups of areas of novelty were defined because there is substantial interest in them by many Member States.

3.1. SITING

3.1.1. General areas of novelty

Layouts for some EIDs can include a fewer number of smaller buildings compared with typical WCRs. This results in a reduced plant footprint, which may provide flexibility in siting. On the other hand, some EIDs may have a larger footprint than comparable WCRs due to additional equipment needed for operation or cogeneration. An example might be an MSR with on-site reprocessing of liquid fuel.

Some EID layouts include seismic base isolation systems to facilitate deployment in high seismicity areas, and some use artificial earth embankments that ‘bury’ the safety buildings to protect them. Significant embedment of safety structures below grade, introduced in some novel designs, may make them less vulnerable to external human induced hazards, including aircraft crash, and may reduce measures needed to offset seismic hazards in above grade structures. On the other hand, achieving significant below grade construction may be difficult or more costly in rock sites, and it could increase the risk of flood hazard. This is, therefore, a trade-off to be considered during site selection.

The general site characteristics and potential locations for EIDs vary considerably, with many having similar characteristics to WCRs. However, some developers consider that their designs have a small source term, low risk or low extent of radiological consequences from accidents owing to the use of advanced safety features, such as passive safety systems. Provided that the regulatory body can be satisfied, such EIDs could be sited within industrial parks or close to population centres.

The passive safety features that have been introduced to many EIDs may make them less vulnerable to loss of off-site power or station blackout. In addition, many designs also claim less reliance on water sources for power operation. Such features make designs less dependent on more traditional forms of ultimate heat sinks (UHSs) (e.g. large water body) which, as a result, may provide greater flexibility in site selection; for example, these designs might be sited in regions where water is not abundant. The rejection of decay heat into a medium with less or no dependence on water (e.g. the air, the ground) could be a new factor in the siting of these EIDs. There are also designs that are intended to be located in remote or extreme climate areas (e.g. in the Arctic).

Although not very frequently, WCRs have already made use of concepts such as siting away from water bodies, use of seismic base isolation and building embedment. It follows that, from a general perspective and excluding TNPPs (see below), there is no aspect of siting of EIDs that is essentially new. Instead, areas of novelty in the siting process might be found in the way that advanced safety features introduced in many of the new designs are credited to reduce the effort in site characterization and to streamline the site licensing process. From this point of view, and acknowledging that this requires further study, it may be that the safety requirements for WCRs might sometimes exceed what is necessary for EIDs. At the same time, addressing this topic is not just a siting issue, since it depends on the interaction of the site conditions with the plant design and the technology of the reactor. Conditions are different for each specific site–installation combination. There is the possibility of optimization of site–EID interactions that could reduce the effort in site characterization and relevant licensing aspects.

EID concepts are introducing safety provisions designed to potentially reduce off-site radiological consequences in the event of an emergency and, consequently, may help to reduce the size of the emergency planning zones (EPZs) and the level of emergency planning compared with existing ‘large’ NPPs. If this ambition is realized, an important challenge for the siting process will be to develop and justify acceptance criteria for siting at locations next to or within densely populated areas.

3.1.2. Areas of novelty common to small size, multiple modules or modularity

No additional areas of novelty specific to SMRs were identified.

3.1.3. Areas of novelty specific to non-water cooled reactors

High temperature reactors, cooled by gas or molten salt, could provide process heat and, by so doing, substitute for fossil fuels in some heavy industries. Such reactors, therefore, may be sited near an industrial plant.

3.1.4. Preliminary areas of novelty specific to transportable nuclear power plants

Marine based or transportable reactors do not have a site during navigation or transportation. However, except for very low power reactors, a process for site selection and characterization is needed for the locations at which these reactors intend to be placed and connected to an electric power or heat distribution network. The process has many aspects in common to the site selection and characterization process of a land based reactor.

3.2. DESIGN

3.2.1. General design approach

3.2.1.1. General areas of novelty

The general approach to design safety for EIDs may be different from the one used by current WCRs, and the difference can be substantial where the barriers for preventing radioactive material from being released to the atmosphere and the implementation of defence in depth (DiD) involve novel concepts and safety provisions. In addition, EIDs may introduce new hazards and postulated initiating events (PIEs), while other hazards and PIEs associated with WCRs may not apply. For example, EIDs make extensive use of passive design features and systems, allowing the performance of safety functions to require little or no electrical power. Thus, passive heat removal systems may rely solely on natural convection, possibly after the opening of some valves. At the present time, however, there is a scarcity of information and a lack of proven experience on PIE identification for EIDs. In addition, while the small size and low power of SMRs and the low power density of some larger EIDs will certainly facilitate the introduction of passive features and systems, there is often insufficient experimental support and experience to fully justify the claims. There will also be differences in important parameters, such as reactivity coefficients and the change in fuel power rating with burnup, and anomalies and hazards associated with the inventory of fission products in the core. In addition, some designs claim practical elimination of some plant states that were typically considered in the design of WCRs; however, many of these claims have yet to be fully justified.

Some EIDs are planned to be located at remote sites, and the developers claim that off-site support is not needed for a prolonged period, even in the most severe accident scenarios. It follows that, compared with operating WCRs, there will be differences in the minimum required autonomy (when there is no off-site support) and the coping time available prior to the delivery of off-site support.

EIDs can offer other uses of NPPs in addition to generating electric power, such as cogeneration of heat for industrial purposes. The cogeneration facility may pose some hazards to the associated NPP, and vice versa. For example, explosion hazards from a hydrogen production plant that is linked to an NPP. Care has to be exercised to eliminate or minimize such hazards so that the risk is kept at an acceptable level.

3.2.1.2. Areas of novelty common to small size, multiple modules or modularity

SMRs may include several modules at a single site, and there may be some shared systems, interactions or other dependencies between the modules that may include or affect safety systems. Where this is the case, safety requirements will need to be framed so that any sharing, interactions or dependencies do not impair the safety systems' ability to perform their functions.

In the case of a multi-module plant (such as an SMR), the implementation of DiD for each module is part of the overall implementation of DiD for the multi-module plant. Levels of DiD and their associated provisions may be assigned to the module or plant (or even site) domain, and provisions for the same level may be foreseen in the module or plant domain. Several SMR designs have passive safety features that are shared between modules. An example is shared external pools for passive (decay) heat removal. The expectation is that passive safety features will allow less redundancy or even diversity[12] than would be expected in a standard WCR and, indeed, such features may already be installed with this intention in some WCRs. In such a multi-module plant, the hazards to be considered will include those that are internal to the plant but external to a module and those hazards that propagate from one module to another. In addition, a PIE could affect several modules at once. These possibilities are areas of novelty and, where modules share common safety provisions, this will introduce an additional layer of complexity that may not currently exist in large NPPs.

[12] Redundancy (and diversity) is related to reliability, and whether it is required or not depends on the role of the system.

For SMRs, the construction and the commissioning of a new module may occur while other modules are already in operation. In other words, work in progress on the new module may impact the operation of module(s) already commissioned and operating; care is to be exercised to avoid such impact.

For SMRs, it appears necessary to define the accident level on one module that will trigger a scram (and maybe other preventive or mitigating measures) on other modules to prevent any propagation of the accident. There is also the possibility that several modules experience an abnormal event simultaneously.

The areas of novelty mentioned above are prevalent in the design of SMRs, but they may also apply to EIDs in general.

3.2.1.3. Areas of novelty specific to non-water cooled reactors

The barriers that retain fission products could be different from those in conventional WCRs, and hence the provisions implemented for each DiD level may differ substantially as well. For example, HTGRs and liquid fuel MSRs do not use fuel cladding. HTGRs use coated fuel particles with multiple barriers. For MSRs, the aim is that even under postulated accident situations, the liquid fuel of MSRs should retain a substantial fraction of radionuclides because of its inherent chemical characteristics. This also applies to solid fuel MSRs with salt as coolant and to SFRs, because they use sodium as coolant, and iodine is retained by sodium in accident conditions owing to the chemical properties of sodium. Lead in an LFR may also retain some fission products. However, salt, sodium and lead do not retain all radionuclides. In particular, fission noble gases are hardly retained at all. In the case of liquid fuel MSRs, fission gases and volatile fission products may be continuously removed from the fuel during normal operation and stored outside the core. Consequently, there would be a reduced inventory of fission products within the core, and those that remained would be more likely to be retained in the salt in the case of accidents impacting the core. The capability of sodium, lead and salt to retain certain fission products depends on their physical (pressure, temperature) and chemical (e.g. redox potential) characteristics.

Compared with WCRs, EIDs may differ in the way that the fundamental safety functions — confinement of radioactive material (and containment of releases), control of reactivity and heat removal — are achieved and demonstrated. For example, some developers aim to achieve the fundamental safety functions via the inherent characteristics and passive features of their EIDs. One example is HTGRs that use coated particles combined with adequate design measures, such as limitation of power and power density, which could provide passive decay heat removal (DHR) — that is, no fuel damage in case of complete unavailability of active DHR systems.

Some EIDs implement a novel approach to the third and fourth levels of DiD compared with conventional WCRs. In particular, the concepts or the characteristics of ‘damage of the reactor core’ and ‘severe accident’ are different than those used for conventional WCRs. Furthermore, for some EIDs it is claimed that severe accidents have been precluded owing to the reinforcement of design measures to prevent core damage from anticipated operational occurrences (AOOs), design basis accidents (DBAs) and design extension conditions (DECs).[13] For the same reason, the scope and definition of the categories of plant states that are considered in the design, and in particular the DECs, can be different, and ‘design extension conditions with core melt’ may not be applicable to some EIDs.

For MSRs that contain all the fission products, including fission gases, in the core region, the severe accident concept used for existing WCR technologies remains useful, as there is still a concept of a structural core — although the definition of a severe accident typically used for WCRs may require using a surrogate accident definition if core damage (e.g. core melt) can be precluded by design. Another example of this preclusion is the case with modular HTGRs using TRISO particles.

For MSRs that use circulating liquid fuel, the salt fuel is normally liquid, and a ‘molten core accident’, defining a severe accident for WCRs, is obviously not relevant. Further, it seems possible by design to reduce the probability of some energetic phenomena (such as hydrogen explosion, which could be physically possible in WCRs in the case of a severe accident). As with HTGRs, therefore, it seems likely that an alternative accident definition will need to be established.

Non-water cooled EIDs may introduce some new or different safe plant states. Examples of potential alternative safe states for MSRs are the following:

(a) Hot low power steady state operation with heat rejection to the normal heat sink or to the available emergency heat sink if the normal heat sink has been lost;
(b) For liquid fuel reactors, the level of fuel salt in the reactor vessel is below the level needed for criticality (i.e. passively or actively draining fuel salt out of the reactor into a holding tank).

[13] The postulation of severe accident conditions is required by the deterministic safety approach in the safety standards, whatever the reliability of the DiD levels or the probability for such conditions to occur is. See the discussion on severe accidents and DiD in Section 4.

As mentioned above, there are some PIEs and failure modes of EIDs that are not relevant to current WCRs. Examples of new PIEs and failure modes for non-water cooled EIDs are the following:

(a) SFR: Fuel assembly flow blockages, sodium leaks and combustion, steam generator tube rupture (SGTR) causing sodium–water reaction, sodium freezing and air ingress into the cover gas.
(b) LFR: Fuel assembly flow blockages, lead leaks, lead freezing and SGTR (liquid metal–water reaction).
(c) HTGR: Air ingress and water ingress potentially causing graphite oxidation or graphite fire. Nevertheless, pure graphite is not very flammable. Developers claim that graphite fire may be precluded if high quality nuclear grade graphite is used.

Similarly, there are some phenomena that are specific to non-water cooled EIDs; for example the following:

(a) SFR: Sodium fire, sodium–water reaction and sodium–concrete interaction.
(b) LFR: Lead-water reaction and lead–concrete interaction. For an LFR using lead–bismuth eutectic, there is polonium-210 production in the coolant, requiring containment or confinement measures.
(c) MSR: Tritium may be generated in certain types of coolant salt, as well as inherent release of fission gases from the salt, which requires dedicated collection of the gases and associated containment measures.

Fire and explosion hazards due to the use of novel materials (e.g. coolants, fuel structural materials such as grids) and additional flammable, explosive or combustible materials may also be introduced by non-water cooled EIDs, such as sodium fires. Even though measures are expected to be taken for these EIDs to avoid undesirable interactions, these potential interactions are mentioned here for completeness, since they are areas of novelty. Mitigation of fires and explosions and their effects, including heat and smoke venting, may be different from parallel measures deployed at WCRs. In particular, interaction of extinguishing substances (such as water, gas, powder) with specific materials introduced by these EIDs may cause potentially hazardous chemical reactions. In addition, systems housing metallic coolants, either for storage or treatment, and systems for fuel storage are potential ignition sources. In general, an EID can have novel primary, or even secondary and tertiary, systems that use new coolants. It is the new types of fluid and material anywhere in an EID that can be of concern.

Regarding internal flooding hazards, there may be floods or spills from the reactor coolant system (RCS) of a non-water cooled EID, and their effects and mitigation measures may be different from those in the case of large stationary WCRs.

Similarly, hazardous substances, such as toxic materials, may be produced by non-water cooled EIDs during normal or accident conditions. Examples are sodium oxides, which can be released after a sodium leak and sodium fire, and lead.

The mechanisms that influence the ageing and failure modes of the SSCs of EIDs will be significantly different from those of WCRs because of, for example, high operating temperature and the chemical properties of the coolant.

The introduction of non-WCRs will generate new challenges with respect to the radiation protection of workers. For example, pebble bed HTGR designs can generate graphite dust that is radioactively contaminated; the lead coolant of LFRs can contain silver and antimony, which are difficult to remove, and their activation products can have high dose rates. For liquid fuelled MSRs, the fission gases are naturally released during normal operation, and then possibly located in specific areas outside the core, which also are challenges for the radiation protection of workers.

In the SFR and LFR designs, there are loops filled with liquid metal that are used for DHR from the primary circuit to the environment. The heat exchanger is located at the top of the building to improve the natural circulation capability. This heat exchanger may need to be protected against lightning, as is the case for other metallic structures located high on the building.

3.2.1.4. Preliminary areas of novelty specific to transportable nuclear power plants

TNPPs also have new PIEs that are specific to their design, and in the case of floating TNPPs, to their aquatic environment. A floating TNPP may use steel instead of concrete for its structures.

3.2.2. Design of the core

3.2.2.1. General areas of novelty

An area of novelty common to many EIDs is the use of advanced fuels. Examples are uranium/plutonium nitride and molten salt, all of which are still in development. Design information in terms of analytical and experimental evidence may not be available for all the new fuel and structural materials, so it may need to be generated.

In some designs, there is an increase in the ^{235}U enrichment of the fuel to almost 20%. Some new fuel designs proposed include ceramic fuel that can be produced in a non-pelletized design and new cladding alloy material such as Cr/Ni alloy, which is an example of accident tolerant fuel (ATF) that is much less susceptible to corrosion or high temperature oxidation than standard WCR fuel. On the other hand, these novel fuels may have different failure mechanisms and consequences, with the corresponding considerations on other aspects, such as short and long term fuel behaviour, fuel handling, fuel cooling and fuel cycle. All the fuel and material degradation aspects, including various corrosion mechanisms, need to be fully understood, so they could require detailed examination to demonstrate that safety is not impaired.

Some specific designs introduce graded axial enrichment to reduce the axial power peaking caused by coolant boiling. Additionally, hollow annular pellets may be used in the upper elevations of the fuel rod to reduce axial differences of burnup rate.

Regarding control of reactivity, most EIDs aim to improve safety (compared with current WCRs) with respect to accidental reactivity transients. Improvements may include an internal control rod drive housing, which removes the potential for a control rod ejection accident; burnable absorbers instead of soluble boron in the coolant; and the use of a boron carbide emulsion injection system as an emergency shutdown option.

Additional automation and remote actions in normal operation and during activities such as maintenance and inspection may be necessary. New remote techniques need to be developed considering the new constraints (e.g. limited access to the structures to inspect, high temperature, opacity of the coolant).

Some designs aim to incorporate additional elements of the fuel cycle into the reactor operation, such as separation of insoluble fission products and continuous adjustment of the fissile content of the fuel salt by on-line fissile material addition (burner reactor) or removal (breeder reactor).

3.2.2.2. Areas of novelty common to related to small size, multiple modules or modularity

For water cooled SMRs, the main areas of novelty are related directly to fuel and core design, including the modifications in geometry and size/scale (e.g. length) and the related reduction in core power density.

3.2.2.3. Areas of novelty specific to non-water cooled reactors

(a) Sodium cooled fast reactors

Similar to WCRs, the key areas of novelty in the fuel matrix of SFRs concern the fuel type and enrichment. Besides uranium oxide pellets, their fuels include MOX, metal alloy, carbide and nitride fuels.

The phenomena that may lead to cladding failure in normal operation are very similar to those that arise in WCR operation. Cladding failure may occur owing to irradiation swelling of fuel, fuel cladding mechanical interaction, corrosion, flow induced vibration and fretting, all of which may also be found in WCR fuel. Cumulative creep damage due to internal pressure and high temperature may also occur in either design but is more prevalent in a fast reactor environment, given, in particular, the higher temperatures and greater neutron fluence. The consequences of clad failure in an SFR may be more severe than in a WCR because sodium and fuel oxide may chemically interact. This is the reason why such failures need to be quickly detected and monitored and in some SFRs, they will initiate a reactor scram.

In evaluating accident conditions, a key design aim is to prevent failure of fuel cladding. In general, the failure mechanisms found in SFRs are similar to those seen in WCRs. They include overheating clad rupture, fuel or clad melting and fuel–cladding mechanical interaction. While there may not be chemical interaction between cladding and oxide or carbide fuel in normal operation, the elevated temperatures expected during accident conditions could cause metal fuel and steel cladding to combine to form a eutectic. This could constitute a novel failure mechanism for metal fuelled reactors. In the case of an overpower transient in an SFR using oxide fuel, fuel pellet melting in the fuel pin may damage the cladding and cause the ejection of molten fuel.

The following novelty considerations related to reactivity control, and therefore to the fuel and core design performance, are identified for an SFR:

(i) There is a reactivity effect associated with any core geometry change in fast spectrum reactors. A compacting motion of fuel subassemblies causes positive reactivity insertion.
(ii) SFRs have active control rod operational systems, but their design is different from those of WCRs. For example, system actuation may respond to different core variables or measurements.
(iii) In general, an important objective of SFR design is to prevent sodium boiling, which may cause positive reactivity insertion and clad melting. This can be achieved by the design features indicated below with the help of a large temperature margin to coolant boiling. The impacts of positive sodium void reactivity arise during core damage sequences. Although positive sodium void reactivity may cause progressive core melt, in-vessel retention of degraded core materials is achievable by limiting positive sodium void reactivity. In general, sodium void reactivity of SFR type SMRs is small or

even negative. This is sufficient to prevent core damage or to achieve in-vessel retention.

(iv) Active reactor shutdown systems such as rapid control rod insertion are employed (in general, two independent and diverse shutdown systems are provided).

(v) Inherent negative reactivity feedback is a feature of core neutronic design and strongly limits the possibility of power excursions through, in particular, the Doppler effect and thermal expansion of the fuel and core.

(vi) Some designs have incorporated a movable reflector that is claimed to be one shutdown measure.

(vii) Passive reactor shutdown has also been proposed, such as Curie point electromagnets, hydraulic suspension rods and gas expansion modules.

(b) Lead cooled fast reactors

This description of core innovations for LFRs is brief because it relies on similarities and differences compared with SFRs.

An LFR fuel matrix may have the following features: metal fuel, UO_2, MOX; mixed nitrides and minor actinide-bearing fuels may be used. In addition, some designs propose the use of high assay low enriched uranium (HALEU) with enrichment of up to 19.75%.

For LFR designs, as for SFRs, there is no eutectic formation between the steel cladding and the fuel if oxide fuels are used. Eutectic formation may potentially occur with metal fuels.

For LFR fuel claddings, the main failure modes are similar to those of a WCR, but there are the following areas of novelty:

(i) The corrosion mechanisms are completely different from those found in WCRs;

(ii) The considerations related to erosion need to include liquid embrittlement;

(iii) The high boiling point of the coolant eliminates reactivity insertion due to coolant voiding.

Developers claim that protection against corrosion by lead may be achieved by maintaining a consistent oxide layer on components exposed to the coolant. This type of protection is specific to LFRs and requires dedicated and highly reliable control of coolant chemistry.

Accident phenomenology that applies to LFR fuel includes overheating clad rupture, fuel melting, corrosion and fuel–cladding mechanical interaction. No critical heat flux concerns due to the high boiling point of the coolant were identified.

The following areas of novelty related to reactivity control, and therefore to the performance of the fuel and core design, are identified for LFRs:

(i) Inherent negative reactivity feedbacks are characteristic of the core neutronic design and they strongly limit possible reactivity excursions;
(ii) Similar to WCRs, there are active control rods, but their design is different from those of WCRs (usually, one fuel assembly has one control rod);
(iii) There are active reactor shutdown systems, such as rapid control rod insertion (in general, there are two independent shutdown systems);
(iv) There are also buoyancy driven shutdown rods, which may be activated passively.

Other differences compared with current WCRs are higher operating temperatures and a fast neutron flux.

(c) High temperature gas reactors

For HTGRs, the primary areas of novelty are the high operating temperature and the use of TRISO fuel (present in pebbles or compacts/prismatic blocks) consisting of multilayer coated particles, with UO_2 or uranium oxycarbide fuel kernels. TRISO fuel can be used in concepts other than HTGRs (e.g. some MSR concepts use TRISO fuel and molten salt as coolant). The use of plutonium and thorium oxide fuel material may also be considered.

The fuel matrix design is clearly different for the HTGR compared with current WCRs. In normal operation and accident response, the HTGR design features have the goal that material integrity limits for structures, systems and components that are important to safety are not exceeded to preserve the effectiveness of the safety functions. For example, the reactor is designed so that in all conceivable conditions, the maximum fuel temperature does not exceed the design limit. To this end, design features might include a slim core, low power density, negative temperature feedback on reactivity, large temperature margins, passive removal of decay heat and a large thermal capacity; the large thermal capacity is derived from graphite (graphite of the fuel pebbles or prismatic blocks, as well as the surrounding (or central) reflector). Given this approach, it is claimed that the HTGR design eliminates many of the key failure mechanisms of concern that might be found in a severe WCR accident (i.e. fracturing, melting, and eutectic formation of zirconium alloy cladding), as well as exothermic high temperature steam oxidation and the resulting cladding embrittlement.

Considering the integrity of the fission product barrier layers of the TRISO fuel, some new phenomena may be involved in barrier failure, such as kernel migration due to temperature gradients; chemical attack of coating layers by metallic fission products; and, for the pebble bed design, abrasion, which could affect a single or multiple elements and be exacerbated by mechanical damage during refuelling. Local critical heat flux and dry-out phenomena are not applicable, given the specifications of the gaseous coolant.

Design evaluation will include consideration of thermal, chemical and other effects that could cause TRISO particles to fail under accident conditions. The following safety function considerations are related to reactivity control and therefore to the fuel and core design performance of HTGRs:

(i) Inherently negative temperature feedback on reactivity via, for example, the Doppler effect.

(ii) The inherent control noted in (i) is supported by a control rod system for the purpose of actively putting the reactor into a specific low power or shutdown state:
   — For pebble bed designs, control rods are in the side (outer) reflector;
   — For prismatic designs (annular core with inner and outer reflector), control rods are placed in the inner or the outer reflector or in both.

(iii) For pebble bed designs, the on-line pebble removal capability is an additional means to shut down the reactor.

(iv) Shutdown in some HTGRs is supported by inherent negative temperature feedback under conditions of anticipated transients without scram. The control rod system may be supplemented by a reserve shutdown system (small absorber sphere system).

For HTGRs, the hot gas temperature in normal conditions is higher than the acceptable temperature for the structures of the primary system, which is pressurized. Therefore, in case of loss of the heat sink, it might be necessary to stop the coolant flow and depressurize the reactor to limit the hot flow in contact with the pressure boundary. This concerns all the EIDs where the hot temperature is higher than the acceptable temperature of the primary system structures. However, a reactor shutdown by itself may not efficiently mitigate this event, because the hot temperature does not decrease quickly after the shutdown owing to the high thermal inertia of the core graphite. This condition may lead to a new safety function for limiting the heat transport of hot gas towards the primary system structures.

For pebble bed HTGRs, fuel operations are made during reactor power operation, without a need to shut down the reactor.

(d) Molten salt reactors

MSRs come in a variety of design concepts that use either solid fuels with salt coolants or liquid fuels. These different configurations of MSRs introduce specific approaches to safety. The developers of MSRs using solid fuel claim that the safety approach is similar those used by HTGRs, given that both utilize fuels that are tolerant to high temperature. The discussions on MSRs in this and other subsections are an overview of these innovations.

The two main types of MSR are defined by the physical state of the fuel — solid or liquid — and this has the main impact on safety. MSR concepts might utilize natural uranium (^{235}U), synthetic plutonium[14] originated from irradiation of natural uranium (^{238}U), or synthetic uranium (mainly ^{233}U) originated from irradiation of natural thorium (^{232}Th) as fissile materials. These may or may not be combined with ^{238}U or ^{232}Th as fertile material.

In MSRs with solid fuel, the fuel is in fuel elements (e.g. pins or coated particles) and is distributed in the core at positions that are static or that change only slowly. A non-fuelled molten salt is used as the coolant. Accordingly, the core composition during operation is not homogeneous, and could require a fuel element loading and unloading plan, which might be periodic (i.e. during shutdown states) or continuous (i.e. during power operation). This configuration is similar to that of innovative non-WCR NPPs (e.g. SFRs, LFRs, HTGRs) with some key differences (e.g. periodic or continuous refuelling).

In MSRs with liquid fuel, the fuel is in a liquid salt solution. Depending on the design, the fuel could be: (i) dissolved in the molten salt coolant, which typically circulates from the core to the heat exchangers and back to the core (in this case, the molten salt is both the primary heat source and the coolant); or (ii) located in tubes within a core that is cooled by a non-fuelled coolant (e.g. a liquid halide salt).

Molten salts (as fuel or coolant) are mainly based on fluoride or chloride mixtures of alkali halides as the solvent, with the corresponding heavy metal halides as solute. Minor constituents are the halides of the fission products and some additives.

In the MSR concepts using liquid fuel, the fission products that are not retained in the molten salt (mainly noble gases and metallic products) are distributed in specific areas (e.g. tanks containing noble gases, metallic products deposited in the heat exchangers).

In relation to reactivity control, the general point can be made that where frequent at-power refuelling is possible (e.g. with pebble bed or liquid fuel), a

---

[14] The Pu isotopes used depend on the neutron spectrum. ^{239}Pu is the major fissile isotope, and all Pu isotopes are fissile with fast neutrons.

reactor can operate with minimum excess reactivity, and this may allow fewer reactivity control devices to be installed. Regarding reactivity adjustments, MSRs with two types of fuel content can be distinguished. For MSRs with constant fuel content[15], in some design configurations, the reactivity variation caused by the change of the fuel composition during operation between two refuelling events is compensated by systems associated with movable parts (e.g. control rods, rotating reflector drums). Minor burnup compensations may be provided by core temperature adjustments for a short period of time. On-line fuelling is another method used in some designs to compensate for fuel burnup (e.g. in pebble bed HTGRs) and to keep the fuel content unchanged.

For MSRs with variable fuel content, there is the possibility of adjusting the fuel composition, such as the fissile content. This possibility is offered by some liquid fuel reactors and some solid fuel reactors such as pebble bed reactors (PBRs). This adjustment can be made periodically, as the salt composition change between two successive adjustments is compensated naturally by small changes in the criticality temperature of the fuel (i.e. the mean fuel temperature beyond which the core is subcritical). Removing and adding small fuel volumes is possible without a need to shut down the reactor.

The high temperature of the MSRs (compared with WCRs) and the radioactivity of the liquid fuel may require automation and remote actions in normal operation, during maintenance and during emergency conditions. This needs to be considered early in the conceptual design.

3.2.2.4. Preliminary areas of novelty specific to TNPPs

In general, the novel design changes for the fuel and core of water cooled marine based NPPs may be characterized as similar to those in land based NPPs. However, the core of a marine based TNPP may experience substantial inertia forces due to the action of waves and wind on the ship or barge on which the reactor is located. This will have impacts on flow phenomena in the core, such as the onset of critical heat flux. It will also impose forces on core components and will particularly impact reactivity control devices such as control rods. In extreme cases, after the TNPP has been installed at the deployment site and is operating, it might even capsize and, for example, gravity driven control rods could fail to insert or fall out of the core if not secured. Such issues are novel in core design and safety cases.

---

[15] The fuel content is never constant for an operating reactor owing to the fission reactions. The term ‘MSRs with constant fuel content’ is used herein to emphasize the difference from ‘MSRs with variable fuel content’.

3.2.3. Design of the reactor vessel and the reactor coolant system (primary circuit)

The EID systems considered in this section are the reactor vessel and the systems designed to maintain adequate cooling conditions for the core in operational conditions. The systems to remove core decay heat in accident conditions and the purification systems of the primary coolant are considered in Section 3.2.4.

3.2.3.1. General areas of novelty

EIDs may use passive safety systems for cooling, especially with respect to DHR.

3.2.3.2. Areas of novelty related to small size, multiple modules or modularity

For integral design SMRs and pool type reactors, non-isolatable piping for the primary coolant outside the reactor vessel or pool is substantially reduced compared with WCRs and might even be eliminated by design. For most reactors of this type, decay heat is transferred to a secondary circuit via heat exchangers located within the primary coolant boundary, thus eliminating ex-vessel coolant piping breaks.

In an SMR of the type ‘integral PWR’, no large break loss of coolant accident (LOCA) occurs, because large bore piping is not used. A water cooled SMR with a pool design may practically eliminate a LOCA if the water level after the LOCA is above the top of fuel for all possible LOCAs. This has implications for the design and capacity of safety provisions such as the means of preserving the coolant inventory to ensure core cooling. In contrast, the in-service inspection of some components important to safety may not be feasible in an SMR with a high degree of integration of SSCs.

For cores having low decay heat, the transport of this heat to the ultimate heat sink may be carried out by passive means.

3.2.3.3. Areas of novelty specific to non-water cooled reactors

The structural integrity of reactor coolant systems may be challenged for non-water cooled EIDs owing to the high temperature of the coolant and, for MSRs and LFRs, because molten salt and lead, respectively, are corrosive. To cope with creep and creep fatigue damage of steel structures such as the reactor vessel, temperatures need to be limited to an acceptable range. Design measures are also provided to adequately manage corrosion. This will include the exclusion of undesirable impurities, such as oxygen and moisture, from chemically reactive coolants (metallic coolants are kept in inert atmosphere) and from the helium primary coolant of an HTGR. These issues may not be confined to reactor operation but may extend to storage and treatment systems for metallic coolants, molten salt and fuels. Corrosion of components (especially important are those components comprising barriers to the release of radioactive material) caused by the chemical characteristics of lead and, in particular, liquid fuel salt is an important new feature.

The key areas of novelty for non-WCRs are the following:

(a) Pressure: High pressures challenge the integrity of the reactor pressure vessel/reactor pressure boundary and are a driver for potential radioactive release in WCRs. Some EIDs operate at low pressure (e.g. SFRs, LFRs, MSRs), which may allow accidents caused by high pressure or loss of pressure to be discounted. In addition, EIDs operating at low coolant pressures do not require depressurization of the reactor coolant system to reach a safe shutdown state. On the other hand, pressurization under accident conditions is a possibility, so design provisions to limit the system pressure are still necessary.
(b) Coolant inventory: Metallic coolant EIDs (SFRs and LFRs) and solid fuel MSRs need not be equipped with a coolant injection system. The coolant inventory necessary for core cooling is maintained by a backup structure such as a guard vessel. For HTGRs with TRISO fuel, it may not be necessary to maintain the core coolant inventory.
(c) High coolant temperature: Some EIDs operate at high temperatures (e.g. SFRs, LFRs, HTGRs, MSRs), which may have undesirable consequences for susceptible materials and components.
(d) Selection of materials: Materials such as carbon steel, stainless steel and graphite are exposed to the operating conditions of EIDs, such as neutron flux (SFRs, LFRs and some MSRs) and aggressive coolant chemistry (LFRs and MSRs). On the other hand, new materials are and will be available, such as quaternary alloys (more resistant against oxidation), silicon carbide cladding (not new, but it can withstand higher temperatures), chromium coated cladding (further increases the pellet–cladding interaction margins) and chromium doped pellets (could reduce the release of fission gases).
(e) Nature of the primary coolant: Examples of primary coolant are gas and liquid metal. There is the potential for release of toxic substances for some EIDs (e.g. SFRs, LFRs) if there were to be a coolant leak. In addition, the high temperature of EIDs (SFRs, LFRs, MSRs and HTGRs) and the opacity

of metallic coolants (SFRs and LFRs) may affect some in-service inspection activities and visual observation.

Further, there are significant innovations related to the provisions for the shutdown and cooling functions in MSRs. For example, many liquid fuelled MSRs have the capability to drain the fuel salt into tanks designed to remove the decay heat while maintaining the fuel salt in a subcritical state. The loss of the salt location control would challenge these safety functions, leading to specific design constraints for the involved devices. On the other hand, there is a risk of flow reduction of liquid fuel salt and its potential consequences (e.g. increase of the fuel salt's temperature in case of local or global flow reduction, or power increase due to the higher amount of delayed neutrons in the core in case of global flow reduction).

The reactor vessel in an MSR with liquid fuel performs the traditional role of the fuel element sheathing and cladding but is designed to be more robust with respect to internal and external events that could damage the vessel. Compared with the fuel element cladding of WCRs, the reactor vessel of MSRs sustains lower loadings (e.g. lower pressure and irradiation damage). It could also be monitored during operation and inspected.

There is a very close relationship between fuel and coolant in MSRs with circulating liquid fuel, since the fuel is dissolved in a salt coolant. Hence, some of the characteristics of this kind of coolant are covered in Section 3.2.2.3.

3.2.3.4. Preliminary areas of novelty specific to transportable nuclear power plants

The design and safety analysis for a factory fuelled TNPP may need to address the operational and transport safety cases as a single entity. Such an approach would give confidence that the appropriate provisions have been made to counter hazards, such as external events, regardless of whether these occur at a site of deployment or during transport to and from the site. This has potential impacts on how standards are applied to design and manufacturing. Some aspects relevant to transportable microreactors, which are often non-water cooled, are mentioned below.

3.2.4. Design of the systems associated with the reactor coolant

3.2.4.1. General areas of novelty

No areas of novelty were identified.

3.2.4.2. Areas of novelty related to small size, multiple modules or modularity

No areas of novelty were identified.

3.2.4.3. Areas of novelty specific to non-water cooled reactors

The following areas of novelty are specific to SFRs, LFRs and MSRs:

(a) Natural circulation of the coolant can be utilized for DHR. The high temperature of the coolant allows the use of atmospheric air as UHS, with less or no need for water to transfer the decay heat to the air.
(b) High boiling temperature of the coolant is an inherent characteristic that delays the operation of safety systems for DHR in accident conditions. The high heat capacity of helium also contributes to this delay in HTGRs.
(c) High thermal conductivity and low Prandtl number of liquid metal coolants (SFRs and LFRs) are inherent characteristics that facilitate heat transfer. On the other hand, there may be a risk of freezing these coolants by overcooling. For liquid metal coolants with melting points well above environmental conditions, this risk may be greatest under shutdown conditions and in the range of subcooling transients.
(d) Low saturation pressure under operating conditions means that high temperatures can be reached in the liquid working fluid without the need for pressurization.

Fission gases (e.g. Kr, Xe, He, tritium) are produced in the fuel during operation. In the MSR concepts with non-fuelled coolant (e.g. solid fuel concept), these gases can be kept and accumulated in leaktight fuel elements. This situation is similar to that found in WCRs, SFRs and HTGRs. In the case of fuel failure, a fraction of the fission products could be released into the coolant salt, with gaseous fission products behaving as described for vented fuel below.

In vented fuel MSR designs — that is, in the liquid fuel concept (including narrow liquid fuel elements[16]) — these gases are naturally released because they are partly soluble in the fuel salt but saturate it within a short time during reactor operation, so they gather in the gaseous phase associated with the fuel salt. Some LFR designs may also consider using vented fuel.

Non-water cooled EIDs have novel systems associated with the coolant. The main areas of novelty for MSRs are directly related to chemistry management and control and, where applicable, gas management. The associated coolant

[16] 'Narrow liquid fuel' is a core concept where the fuel is liquid and located inside vented pins, which are cooled by a non-fuelled medium.

cleanup systems are significantly different from those in WCRs in terms of what is removed, how it is removed and what wastes are generated. The degree of control of chemistry conditions will vary from design to design and according to the reactor's operation mode. For example, measures are necessary to monitor and manage corrosion and the formation of solid fissile deposits and to correct the shift in fuel composition. Such measures may range from monitoring provisions to sampling and cleanup systems. At a smaller scale, a chemistry management and control system is also needed for solid fuel reactors.

In vented fuel MSR designs, inert gas (e.g. argon) may fill the empty space above the surface of the fuel salt and fission gases released from the fuel salt will collect here. Extraction, storage and reprocessing of these fission gases is an option for these reactor designs.

To avoid leaks, corrosion control is also necessary in MSRs to support long term reliable operation and confidence in the integrity of the boundaries containing the molten salt. The concept involves new high temperature materials and technologies (e.g. instrumentation) adapted to molten salts. Corrosion may arise from normal chemical changes due to fissions in the case of liquid fuel (predictable) or from external chemical pollution (accidental). There may be several types of technique that are available for corrosion control. The main provision against corrosion is the fuel oxidation–reduction (redox) potential control. Several passive and active techniques are foreseen, whose effectiveness needs to be demonstrated. Corrosion control is also an issue for LFRs and requires monitoring and management of the chemical content of the coolant.

For liquid metal EIDs (SFRs and LFRs), oxidation of the coolant may generate solid oxides, which may disturb the coolant circulation.

Prevention and mitigation of air or water interaction with metallic coolants (e.g. sodium, lead) may require some new ancillary systems, such as for monitoring and cleanup. In particular, chemical control of sodium and lead (especially oxygen levels) is more significant than for WCRs.

Helium leakage and dust transport are novel characteristics of helium cooled HTGRs. Helium leakages in normal operation, especially when the gas is at high pressure and temperature, are unavoidable because of the inert nature of helium. This necessitates specific considerations in HTGRs, such as an adequate helium supply system, monitoring of the helium content and monitoring of off-site releases. Dust is generated in the reactor system during HTGR operation and needs to be managed. Dust deposition and removal can affect effective transport of heat from the reactor core to the secondary side and could have consequences in normal operation, maintenance and accidents.

3.2.4.4. *Preliminary areas of novelty specific to transportable nuclear power plants*

No areas of novelty have been identified under this heading. Transportable microreactors are often non-water cooled. Related areas of novelty specific to non-WCRs are presented in Section 3.2.4.3.

3.2.5. Design of the containment

For the containment system of the reactor, Requirement 54 of SSR-2/1 (Rev. 1) [3] 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."

In WCRs, the containment system comprises a containment structure and its support systems. For non-water cooled EIDs, however, the functions of the containment system may be achieved quite differently. An engineering combination of design provisions (systems) and civil structures may be proposed to meet containment safety objectives. In HTGRs, for example, the confinement of radioactive substances under operational and accident conditions is mainly achieved by the TRISO fuel, which provides a robust barrier; a containment structure, such as the one used by WCRs, may not be required to provide this function. Nevertheless, it is still needed for other functions, notably protection against natural and human induced external events. The requirements and recommendations related to the containment system in the IAEA safety standards do not consider this major novelty, nor do they address situations potentially arising from the EIDs. Nevertheless, an entity, such as the NPP operator, needs to demonstrate that the proposed safety provisions practically eliminate large releases of radioactive material to the NPP's surrounding environment and that other releases are kept below acceptable limits and as low as reasonably achievable. This concept of a confinement function using barriers other than the traditional containment system of WCRs is sometimes referred to as 'functional containment'. As is well known, WCRs have other barriers to prevent the release of radioactive material to the environment, in addition to this containment system.

3.2.5.1. General areas of novelty

In general, designs with completely passive safety systems aim to manage mass and energy release without motive power. Hence, some designs may have less redundancy in systems providing this function.

Underground/submerged installation of reactors, containment vessels or both is an area of novelty, and new kinds of hazard may need to be considered. This design option may need new technologies to inspect the reactor vessel.

Different technical solutions for access to containment vessels also are areas of novelty. The use of airlocks for access to containment may not be relevant for some technologies with steel containment vessels. In addition, small containments where no access is necessary during operation may utilize robustly closed features (e.g. bolted flanges) that do not need to be frequently and rapidly opened.

The layout and configuration of the containment and its associated systems for EIDs may be significantly different from those used in WCRs. The containment in a WCR and its internal components provide the first mechanisms for the removal of airborne radioactive material, since they present a large surface area for deposition. However, for some EIDs, the surface area of the containment vessel is much smaller, and the efficiency of the deposition process may be reduced.

For some innovative reactor designs, the containment itself fulfils the safety function of removing residual heat from inside the containment.

An SGTR is a potential containment bypass event that could lead to a radioactive release to the atmosphere, and it is also relevant to EIDs using steam generators.

3.2.5.2. Areas of novelty related to small size, multiple modules or modularity

The containment for multi-module plants is a novelty that is similar to already existing reactor designs that have a common containment for more than one reactor. Specifically, in some SMR designs several reactors share equipment, such as a common reactor pool. Accident conditions affecting several modules at the same time in the case of multi-module concepts may have a larger impact on the integrity of the containment than conditions affecting a single module.

Designs with containment vessels that are immersed in water pools may not require a ventilation system to be operated to maintain the pressure and temperature inside the containment during normal plant operation.

3.2.5.3. *Areas of novelty specific to non-water cooled reactors*

The containment structure is the ultimate barrier to prevent unacceptable releases of radioactive material to the atmosphere in case of accident conditions in current WCRs. On the other hand, some EIDs, especially HTGRs, apply a confinement concept instead of a containment system by claiming that sufficient retention is ensured by other barriers. Currently, most HTGR designs are not equipped with a containment structure, and where one is present, it may have minor importance for retention of radioactive substances.

For any non-water cooled EID, a civil engineering structure is needed to protect it against external hazards. This structure might be, partially or entirely, the same as the containment structure (where one is present).

Other loadings than those considered for WCRs will need to be considered; for example, high temperature and chemical damage induced by a sodium leak, a sodium fire or a sodium reaction with concrete.

MSRs also use a different approach to barriers for preventing radioactive material from being released to the atmosphere than that used by current WCRs. For solid fuel MSRs with leaktight fuel elements, the molten salt is the coolant and may have capacity for retaining fission products. The fuel element cladding (e.g. pin clad, particle coating) also provides a barrier. If the fuel element cladding fails, some MSR concepts may employ dilution and chemical retention of radioactive material by the coolant. The reactor vessel and the primary circuit would then be treated as a means of providing confinement, as an integral part of containment or both, depending on the design configuration.

For liquid fuel MSRs, fuel salt leakage and fission gas leakage may lead to release of radioactive materials inside the containment. Therefore, a careful design of the confinement features is required (e.g. high leaktightness, collection of leaking materials, preclusion of containment bypass).

The HTGR and MSR designs using TRISO fuel may have high resistance to thermal loadings, inherent capability for reactivity control (e.g. efficient negative reactivity feedback) and capability to passively remove heat during shutdown and accident conditions (e.g. heat radiation through the reactor vessel). Hence, these EIDs may not need an additional containment structure enveloping the reactor system.

Some EIDs (e.g. HTGRs) are not expected to be susceptible to a core melt accident; this will reduce the possibility of any accidental release beyond the primary circuit. In addition, such designs may not require instrumentation for the localization of the molten core in an ex-vessel retention strategy.

Severe accident phenomena in WCRs may result in the release of fission products, hydrogen and oxygen within the containment structure. Some non-water cooled EIDs, such as SFRs, LFRs and MSRs, do not need to accommodate and facilitate a large energy and mass release or a pressure increase inside the containment during accident conditions. In general, the conditions inside the containment in case of an accident may be less severe for some EIDs (e.g. HTGR, LFR, SFR, MSR) than for a WCR. In particular, MSRs with either solid or liquid fuel operate at a lower pressure than WCRs do. This characteristic means that there would be less energy available for the release of radioactive material to the atmosphere in the case of accidents in MSRs than in the case of WCRs, which operate at much higher pressures. It follows that there is also a requirement to preclude, by design, possible energetic reactions between salts and other fluids. For example, a large failure of the MSR vessel leading to the release of a significant amount of fuel salt would, if contact with water was avoided, result in much less severe mechanical loadings on the containment than those that may occur as a consequence of an accident in a WCR. This could reduce the constraint on the containment compared with WCRs.

The large inventory of fission gases collected outside the core region in some MSR concepts represents a special consideration in the design of the confinement barriers. Fission gases need to be managed in specialized collection systems with pressure management and temperature control provisions. Failures of such systems and interconnecting systems to the reactor itself may represent a significant potential release of fission gases and need to be prevented and mitigated adequately by design provisions.

Mass and energy released from chemical reactions during accident conditions in non-water cooled EIDs may pose challenges to a containment in an analogous way that reactions between molten materials and water pose challenges during severe accidents in WCRs. In addition, some non-water cooled EIDs may release other substances to the containment atmosphere. For example, carbon monoxide can be released in the case of air or water ingress accidents in an HTGR; however, the TRISO particles would continue to perform as a confinement barrier, and opening of venting devices would release carbon monoxide to the atmosphere outside the reactor building.

HTGRs may apply the approach of early containment venting in the case of a LOCA (release of helium as a non-condensable gas). HTGR vendors claim that the TRISO fuel is not significantly damaged in a LOCA, because of the inherent characteristics of the reactor and fuel. Moreover, they claim that the radioactive content of the helium coolant is low in normal operation. Assuming these claims to be correct, in a LOCA the reactor building could be vented to the environment without exceeding the relevant radiological criteria.

There may not be large water reservoirs, pressure suppression pool systems or sumps within the containment of non-water cooled EIDs (such as SFRs, LFRs and HTGRs). A containment sump for emergency core cooling may not be necessary, so concerns about clogging of the sump may not be applicable.

However, debris within the containment may challenge the efficiency of passive heat removal. Parameters other than humidity may be relevant for indicating a coolant leak in non-water cooled EIDs.

For liquid metal EIDs (SFRs and LFRs), some developers claim that a degraded core can be retained inside the reactor vessel (in-vessel retention). The containment is required to confine gaseous and volatile radioactive materials that are not retained by the coolant.

Some non-water cooled EIDs (SFRs, LFRs, HTGRs) have alternative provisions for the management of containment pressure and temperature for postulated accidents, since the traditional water spray systems of WCRs are not technologically appropriate. In some cases, developers claim that no special provisions are necessary, subject to a justification and demonstration that the civil structures are sufficiently robust. Such a demonstration for a first of a kind project would be technologically challenging. For WCRs, phenomena that would need to be addressed include radiolysis of the water in the core, radiolysis of the water in the sump or in the suppression pool, degassing of hydrogen dissolved in the primary coolant, releases from the hydrogen tanks used for control of the primary coolant chemistry, and the risk of hydrogen combustion while the steam concentration is decreasing. For non-water cooled EIDs, the phenomena that pose a challenge to containment provisions can be significantly different and need to be derived on a case by case basis to understand their contribution to risk. For example, under certain accident conditions, combustible and potentially chemically reactive materials (such as sodium in SFRs), or even radiologically significant substances, might be produced and need to be mitigated by the design features that support safety functions. In some cases, such as certain MSRs, freezing of the salt (the salt expands) may mechanically challenge the integrity of components in the fluid systems.

For MSRs with vented fuel, provisions are likely to be similar to concepts with leaktight fuel elements, where the chemical properties of both fuel salt and coolant salt may contribute to the function of confinement. Only the fuel salt can contribute to this function in the case of liquid fuel MSRs. On the other hand, this type of MSR, including liquid fuel concepts that implement venting, introduces the need to handle large quantities of fission gases because these gases inherently separate from the fuel salt and are collected in the gas system. Management of these gases introduces the need for extended means of confinement and robust containment provisions to prevent their release under postulated accidents. This need is common to all EIDs using vented fuel.

Likewise, for MSRs with liquid fuel, the use of fuel salt transfer and storage systems (fuelling systems, chemistry cleanup/processing and drain systems) necessitates extension of provisions for confinement and containment. For example, the fuel salt in normal shutdown conditions could be drained. The drain tanks (and the connection to the reactor vessel) need to be treated as a means of providing confinement, as an integral part of containment or as both, depending on the design configuration.

3.2.5.4. Preliminary areas of novelty specific to transportable nuclear power plants

Containments for TNPPs, including floating TNPPs, and containment designs without concrete inside the containment or supporting the containment structures are areas of novelty. In particular, some floating TNPP designs lack concrete structures, so interactions between concrete and a molten core are not applicable.

Another novel consideration is the confinement function during construction and transport of factory fuelled SMRs. In addition, factory fuelled SMRs and transportable SMRs may be constructed and operated at several sites with different SL-2 earthquakes[17].

3.2.6. Design of electrical power systems

3.2.6.1. General areas of novelty

As the safety architecture of some EIDs relies mainly on passive and inherent safety features (e.g. passive heat removal), requirements for an alternating current (AC) and direct current (DC) power supply to support safety and operational systems may differ substantially from current industry practice. Facility location and function will also influence the design. For example, a facility located in a very remote site (e.g. off-grid) may need specialized power supplies and distribution systems to maintain additional systems needed for preserving the asset or for supporting safety. In particular, some developers propose that AC electrical power may not be immediately required to maintain the plant in a controlled state following an anticipated operational occurrence or accident condition. Accordingly, the critical functions performed by the safety systems required for safe shutdown, core and spent fuel assembly cooling, containment isolation and integrity, and integrity of the reactor coolant pressure boundary may be initially supplied only by DC power sources. Hence, there is more emphasis on the design and reliability of DC power systems, including the associated uninterruptible power supplies (e.g. single failure criterion, independence, diversity) to supply the necessary loads to fulfil these functions.

---

[17] An ‘SL-2 earthquake’ denotes the level of ground motion associated with the maximum earthquake to be considered for design.

The power supply from a safety class 1 DC power system may be needed (unless it can be demonstrated by safety analysis that it is not needed) to maintain the plant in a controlled state by initiating passive safety systems and supplying power to I&C systems for monitoring and control purposes. If DC power is supplied by batteries, their ability to supply power would extend over a long period if the power demand was low. Such a situation could apply for some EIDs if, for example, no large loads were required for coping with PIEs, including a station blackout event. Developers consider that AC power systems may not be needed immediately at the onset of an accident and may be manually started later to ensure this continuity (e.g. charging batteries) beyond the nominal DC supply operating time. The goal is to provide power for monitoring critical parameters during operational states and accident conditions, even for plants with inherent safety characteristics, to replenish or restore inventories to passive safety systems required to maintain core cooling and to provide power supply to maintain a safe state for the overall plant for an extended duration.

The safety classification of on-site DC power systems is dependent on the safety functions performed by the supported systems as determined by safety analyses. Regardless of the safety provisions in the facility, appropriate power supplies and distribution systems are necessary to maintain adequate monitoring and control of plant states by the operations staff and to maintain any plant features necessary to support security and safeguards provisions. This includes sufficient power to maintain lighting and environmental conditions in control areas and equipment rooms. Additionally, reliable power supply should be available to any necessary devices that provide situational awareness to plant staff, support any necessary mitigative measures or both. The siting conditions of the facility can have a significant influence on which devices are necessary, considering the event progression grace time and confidence in the performance of safety provisions.

Accordingly, the use of passive safety features is being used by proponents to make a case that the safety of the facility may not immediately depend on on-site or off-site AC electrical power and that the power supply architectures can be simplified. In particular, it is proposed that some facilities may not require a safety class 1 emergency on-site AC power system and that if an emergency on-site AC supply was needed at all, this system would then be classified according to its safety significance. The safety classification process would then be used to determine and justify the engineering design rules to be implemented. This would, for example, be used to determine the size and reliability of backup and emergency power sources. AC power may be needed in the long term for EIDs with metallic coolants to prevent freezing of the coolant. AC power is generally more efficient than DC power for this purpose. In particular, this could be the case when the decay heat is very low and natural convection is no longer possible.

Active forced circulation is then necessary for preventing freezing. However, any such technical proposal will need to be suitably supported with sufficient information providing confidence in plant operating characteristics, taking the human operational model into account. In any case, a DC power supply will be needed for the surveillance of plant status throughout any accident scenario.

Developers also consider that a large standby backup power source (e.g. diesel generator) that can start in a short period of time (e.g. 10 s) and power LOCA loads may not be needed either. An on-site AC power source may be provided to support the plant auxiliaries in the event of an extended loss of an off-site power source. In addition, developers propose that the number of off-site power sources and the composition of the electrical switchyard may be simplified and, accordingly, a single off-site power supply source may be acceptable for an EID employing passive engineered safety features.

The architecture of the electrical power supply system may be determined by the passive nature of systems used for achieving safe shutdown, since these systems may be powered from DC power systems.

Some EIDs (e.g. HTGRs) may be designed to produce process heat supply, for example, for hydrogen production. These EIDs do not generate electric power and, therefore, do not provide electricity to the off-site grid. Nevertheless, an electric switchyard and off-site power connections may be needed to support a controlled safe state.

Some EIDs can be designed to start up from a completely de-energized state without receiving energy from the off-site electrical grid. This is commonly known as ‘black start capability’. This can help the grid to steadily supply power to critical loads and start larger generation units. For an EID to have this capability, an AC power source, such as a diesel or gas powered generator, needs to be located on the site or near the site and be able to start and operate for a few hours in an isochronous mode.

3.2.6.2. Areas of novelty related to small size, multiple modules or modularity

A multi-module plant (such as an SMR) may enable one module (when coupled with the necessary electrical system design features) to supply housekeeping and shutdown cooling power to other reactor modules. This feature enables island mode operations[18] and enhances shutdown DHR functionality. Such a design may also enable one module to supply startup power to other reactor modules.

---

[18] Island mode is an operating mode in which an NPP is isolated from the transmission network (both load and off-site power supply) and operating at a power level sufficient to meet all its housekeeping loads.

3.2.6.3. Areas of novelty specific to non-water cooled reactors

No areas of novelty that are only specific to non-WCRs were identified.

3.2.6.4. Preliminary areas of novelty specific to transportable nuclear power plants

For a TNPP facility (marine or land based), different designs will have different approaches regarding which electrical systems will be dedicated to the transportable portion of the facility and which to the non-transportable portion, which could be a permanent part of the site infrastructure. Due consideration of how TNPP electrical systems interface with the site infrastructure needs to be taken to ensure that power supply reliability targets can be met with sufficient confidence. This is particularly important if the TNPP is sited in a region with unreliable power infrastructure and will inform how self-reliant the TNPP will need to be.

3.2.7. Design of instrumentation and control systems

3.2.7.1. General areas of novelty

Many EIDs have significantly reduced the scope of I&C safety systems or have eliminated the need for I&C safety functions altogether. EIDs using passive features to cope with PIEs may not need the support of I&C safety systems, or the dependence on these systems may be less than that in current WCRs. Therefore, the safety classification of some I&C systems may be assigned according to their safety significance.

Some developers consider that the extensive use of passive design features and systems by EIDs greatly reduces, or even removes, the need for external power, for external control by human operators and for automated I&C systems. See also Section 3.2.6.

Designers are also considering the use of I&C industrial standards and technologies that have emerged in recent years or older technologies that have not yet been extensively used in the nuclear industry. Examples include open platform communications unified architecture, time sensitive network remote input/output and field buses (which have accumulated significant operating experience in other industries) and extensive use of multiplexed communications for class 3 or not classified I&C systems. The use of on-line monitoring to promote condition based maintenance or off-site monitoring is also being considered to assist local operators in prognostics and diagnostics. Evidence from other industries indicates that these activities may promote optimized maintenance.

EID designers are also evaluating options for implementing advanced automation systems to make them locally (on-site) autonomous, with the ability to monitor and make operational interventions from a remote location.

Possible developments in human factors engineering that could be introduced with EIDs include the following:

(a) The use of advanced automation systems. These may produce a more dynamic human–automation interaction, but they will need to consider lessons learned from other industries that use them (e.g. aerospace).
(b) Some advanced designs that, because of their configurations or specific hazards, will likely require the use of robotic systems to support operations and maintenance. This requires new considerations in both the operations and maintenance of such equipment.
(c) Some designs may introduce the use of enhanced operator tools such as wearable human–system interfaces (HSIs) and remote facility support facilities. There are several considerations that will need to be addressed, including interface compatibility and cyber security.

3.2.7.2. Areas of novelty related to small size, multiple modules or modularity

Some SMR designs employ a primary circuit that is fully integrated within the reactor vessel (including steam generators, pumps, pressurizer and control rods), which would require using advanced measurement methods for monitoring of process parameters.

In conventional plants with multi-unit control rooms, typically, a single crew is dedicated to a single reactor, with the possibility that they or someone else is responsible for the balance of plant systems. An SMR may use a smaller staffing model commensurate with a potentially simpler safety case than for a current WCR. In an SMR, the novel proposal may be that multiple modules (reactors) at a single site may be operated by a single team from a single main control room (MCR). This team may be responsible for multiple reactor modules, associated module secondary systems and possibly shared systems.

The design of a multi-module SMR plant often has some plant systems shared by all or some modules. This naturally impacts on I&C architectures, control rooms, control room systems and electrical power systems.

SMRs may use remote monitoring and support centres more extensively than current WCRs. The same SMR design may be used at different sites, so some projects may consider remote centres for monitoring and support in normal operating conditions and under accident conditions.

Maintenance represents an important part of operation and personnel costs, and SMR designs often look for ways to optimize it (e.g. by using on-line monitoring to promote condition based maintenance or off-site monitoring to assist local operators in prognostics and diagnostics). Longer operation cycles and reduced inventory of active components (e.g. valves, pumps) may also need to be considered when addressing on-line and off-site monitoring.

There may be a distinction between the overall I&C architecture of a plant and the overall I&C architecture of individual modules at the plant.

Some possible areas of novelty related to computer security in SMRs are: (i) computer security during transportation to a site of modules that are fully assembled, configured and tested in a factory; (ii) staged construction; (iii) remote monitoring and support centres; (iv) the existence of multiple, quasi-identical units in many different geographical sites; and (v) separation (from a computer security standpoint) between the modules of a multi-module plant.

With respect to human factors engineering, the following aspects need to be considered:

(a) Multi-module plants may have modules in different stages of the nuclear facility lifetime (i.e. construction, commissioning, operation or decommissioning), and multiple operational modules can be in different operational states (e.g. refuelling, outage, AOO, DBA). Hence, the HSI may be different from that of a current WCR in that it includes operational strategies that can be applied when modules are in different operational states or different lifetime stages.
(b) When a control room that is shared between modules and other shared operational support facilities, simultaneous monitoring and control of multiple modules is different to the monitoring and control of a single unit, which is current practice for WCRs.
(c) For HSIs for shared systems, some considerations include: (i) the interdependence, overlap or redundancy of these HSIs raises the possibility of common cause failures (CCFs) that can affect multiple modules; (ii) modification of these HSIs may be a particular issue if there are modules in operation while new modules are being constructed; and (iii) HSIs may be provided to isolate modules from shared systems, as necessary, for maintenance and other activities.
(d) Potential module to module differences may — intentionally or unintentionally — affect human performance, which, in turn, will influence situational assessment and response planning. Intentional differences can be HSI design aspects of module differences that are planned and standardized. Planned and standardized HSI design is likely to be key to supporting the ability of plant personnel to distinguish between modules, particularly as part of outage management, when correct module identification is essential to safety.

(e) Remote control of a module may allow HSIs to be simplified but, if not implemented well, could have the opposite effect.

3.2.7.3. Areas of novelty specific to non-water cooled reactors

No areas of novelty were identified.

3.2.7.4. Areas of novelty specific to transportable small modular reactors

Transportable NPPs may not require seismic instrumentation.

3.2.8. Design of auxiliary and support systems

3.2.8.1. General areas of novelty

Designers of EIDs propose to use inherent and passive features to cope with PIEs and to reduce or eliminate reliance on auxiliary and support systems for safety. The number of these systems may then be reduced, or their safety class classified according to their safety significance in accordance with a graded approach. Examples are given below for various types of auxiliary systems.

(a) Heating, ventilation and air-conditioning systems. Related to the above point on the classification of some support systems of EIDs, if forced ventilation is not required in the short term or to reach a safe shutdown state, both the heating, ventilation and air-conditioning (HVAC) system (or some part of it) and the associated power supplies may be classified according to their safety significance consistently with a graded approach. Other examples of differences in the safety significance of support systems follow.
    (i) Part of the ventilation system in a WCR provides dynamic confinement of potential contamination by ensuring that air flows are always towards the most contaminated rooms. This may not be needed in EIDs if a passive system is installed for maintaining the confinement in accident conditions. For example, plant operators in the MCR need to be protected from accidental releases of toxic or radioactive gases. In an EID this could be assured passively by providing compressed breathable air to the MCR. If the ventilation system is passive or not needed, then the associated chilled water supplies may be passive or can also be dispensed with.
    (ii) Ventilation for the purpose of cooling rooms that contain electrical motors or pumps may not be required if natural circulation can supply adequate cooling in accident conditions.

(iii) In some EIDs, diesel generators or gas turbine generators are installed but with a safety class lower than class 1. Consequently, the HVAC system supporting the diesel generators may not need to be rated as class 1.

(b) Heat transport systems. These are auxiliary systems to remove heat from systems and components at the NPP. They do not include systems for DHR. No areas of novelty were identified.

(c) Lighting and emergency lighting systems. As the designs of some EIDs rely more on passive systems that may not need manual actions during accident conditions, the plant locations where lighting is necessary in accident conditions may be different.

(d) Supporting systems for the emergency power supply and the alternate power source. If an EID is designed with a black start capability (to start up from a completely de-energized state without receiving energy from the grid), the fuel oil storage, air start systems, DC power and other supporting systems must reflect this.

(e) Compressed air system. For EIDs using mainly passive systems for the safety functions, the compressed air system is less frequently used as a supporting system of a safety class 1 or class 2 system. In this case, the compressed air system may have a reduced safety classification in line with the graded approach.

3.2.8.2. Areas of novelty related to small size, multiple modules or modularity

(a) Shared auxiliary and support systems

Where auxiliary and support systems are shared between several modules of an EID, failure of a shared system may affect several modules. Similarly, the commissioning of a shared system or the connection of a new module on a shared system may impact operating modules.

In addition, for multi-module EIDs that increase plant capacity by the installation of additional modules, the loads on auxiliary and support systems will also increase. This increase may not be limited to shared systems, but could extend, for example, to the HVAC system of the building where the shared systems or modules are located. For this reason, the initial design capability of the auxiliary and support systems may consider the supplementary loads that will be produced by the installation of additional modules.

There may also be a shared communication system of a multi-module EID.

(b) Heating, ventilation and air-conditioning systems

In the case of an accident where water in the spent fuel pool of a light water SMR boils, an unwanted buildup of pressure in the fuel building could be prevented by the HVAC system passively evacuating steam using the exhaust line of the ventilation via suitable filters. On the other hand, the dynamic confinement mentioned in Section 3.2.8.1 would not operate and the efficiency of the filters (e.g. iodine trap) may be strongly dependent on the air humidity. Further, in the case of boiling large water pools in SMRs, the volumetric flow rate of air and steam might be larger than the flow rates achievable via a filtered route at pressure differences allowable for the integrity of reactor building openings such as doors. In such circumstances, other discharge routes are likely to be needed.

There may be buildings shared between different SMR modules (e.g. a reactor building sheltering more than one module, one control room for several modules or a spent fuel pool common to several modules). In this case, the ventilation system should be designed to operate during an accident in one module without affecting the remaining modules' capability to reach a safe state.

(c) Overhead lifting equipment (other than fuel handling)

Some SMR designs may require heavy lifting equipment for installation and maintenance. Examples of components that may require such lifting are the reactor pressure vessel of integral PWRs, which includes the steam generators, the reactor coolant pumps and the pressurizer. For factory serviced modules, heavy lifting of a complete module on the site is needed, and the rest is done in the factory. A whole module is lifted into its refuelling position in an SMR design.

In addition, SMR designs may have limited space for a module compartment, so various types of lifting equipment (e.g. forklifts) may be used, in addition to overhead lifting equipment, as the means to move and relocate heavy equipment within the power plant.

(d) System for the treatment of solid waste

As SMRs could be built in large numbers in a geographic area, a single installation for the management and storage of radioactive waste of all the SMRs in this area may be useful. Such an installation would have greater throughput of waste, and therefore would offer the opportunity for the application of advanced processing technology to reduce the environmental impact. This approach would avoid or reduce the number and complexity of systems and facilities for the management and storage of radioactive waste at each NPP site. It may also apply to liquid and gaseous wastes.

(e) Supporting systems for the emergency power supply and the alternate power source

Current WCRs satisfy the recommendation that each emergency power source should be provided with its own completely independent supporting systems. Experience from existing WCRs has shown the need to anticipate, prevent and mitigate CCFs and single failures that can impact multiple power sources simultaneously. This is also applicable to the case of an SMR with shared supporting systems.

(f) Demineralized water reserve and associated system

Large PWRs may need a demineralized water reserve to supply the auxiliary or emergency feedwater system for long term residual heat removal via the steam generators. On the other hand, the residual heat removal in light water SMRs may be performed by a system that directly transfers the heat to a water reserve. This reserve needs to be resupplied in the long term.

3.2.8.3. Areas of novelty specific to non-water cooled reactors

(a) Coolant control systems

For HTGRs during normal operation, a helium sampling system is used to measure moisture, chemical impurities and radioactivity in the primary helium coolant. During accident conditions, this system is replaced by the process and post-accident sampling system, which samples gas in the reactor containment. This arrangement is similar to what would be found in a current WCR.

For LFRs and SFRs during normal operation, coolant sampling may be used to measure primary coolant chemistry. For LFRs, control of coolant chemistry is needed to maintain a protective oxide layer on the structures in contact with lead. For SFRs, chemical control of the sodium coolant aims to suppress oxygen levels. SFR developers claim that the chemical control of coolants and cover gases need not be continuously performed, and corrective actions (typically purification and, if necessary, reactor shutdown) may be taken as required. During accident conditions, these systems need not be operated to maintain a safe state. Monitoring the reactor containment during accidents can be important to detect possible radioactive releases from the primary side, which is isolated from the containment.

For MSRs, the chemistry of the salt and cover gas are controlled to limit corrosion of structures and, for liquid fuel MSRs, maintain sufficient solubility of the fissile materials.

(b) Radiation monitoring system

A leak from a steam generator tube in non-WCRs can be detected by various means. These include: for HTGRs, monitoring of moisture in the primary helium coolant; and for SFRs, monitoring of water–sodium reaction products.

(c) HVAC systems

The functions typically delivered by HVAC systems in conventional WCRs (as identified in SSG-62 [39]) may be delivered in EIDs by passive systems designed to be independent from external sources of energy (such as an electrical connection to the grid), such as batteries (for their operating time, usually a few hours) and compressed air tanks (not requiring external power). In the case of a radiological release from an SFR or LFR, the HVAC system may need to be isolated by active devices needing DC power.

For some HTGR designs, the provision to keep a negative pressure in rooms in controlled areas in normal operating conditions may not be needed. In accident conditions also, HVAC is claimed not to be required to support safety functions owing to the passive design concept. A break in the primary system would cause a quick increase of the pressure of the reactor building because of the release of high pressure primary coolant. To maintain the integrity of the reactor building in these circumstances, a specific device may be needed to keep the pressure below the maximum design pressure of the reactor building. This device may temporarily disrupt the dynamic confinement. In some concepts, the dynamic containment may not be foreseen to be restored in the short term, which will be a novelty regarding applicable requirements.

(d) Reactor cavity cooling

Heat transport systems may be used for cooling some structures of EIDs, such as the reactor cavity of an HTGR, SFR or LFR. For some EIDs, the reactor cavity cooling system operates in passive mode in accident conditions. For example, some HTGRs may use a reactor cavity cooling system that is designed to operate entirely in passive mode in accident conditions and, depending on the design, to function during reactor operation by either active or passive means. If the reactor cavity cooling system of an HTGR is operated in active mode whilst the reactor is in operation, a heat transport system, including active components such as pumps, is needed to remove heat from the reactor cavity cooling system equipment.

(e) Power conversion system

Compared with WCRs, non-water cooled EIDs can provide heat with significantly higher temperatures. This provides capability for new applications other than electricity generation. For example, some non-water cooled EIDs can be coupled to a heat storage tank containing hot molten salt. Heat is drained from this tank to run a turbine generator when the power grid requirements are high, and heat is transferred to the tank when they are lower. Other non-electrical uses are also possible (e.g. steam production, desalination).

The challenges associated with these new applications of the heat produced by the reactor need to be identified and managed.

(f) Lighting and emergency lighting systems

As the designs may not need manual actions during accident conditions, the plant locations where the emergency lighting is provided may be different from those in WCRs.

(g) Overhead lifting equipment (other than fuel handling)

The extraction of components from the primary circuit needs to be done under non-reactive gas in some EIDs, such as SFRs and MSRs. For LFRs, each component is extractable from the primary circuit using dedicated lifting equipment, which is strongly dependent on design. Handling is made under non-reactive gas in some designs, and other designs are similar to current WCR technology.

(h) Systems for the treatment of gaseous effluents

As mentioned earlier in this section, fission gases are extracted from some MSRs with liquid fuel and other EIDs with vented fuel. In addition, there is tritium generated in certain types of salt used in MSRs.

Specific measures need to be implemented for the management of polonium (which is volatile) in LFRs. For HTGRs, gaseous radioactive waste comes from the helium purification system; this is unique to HTGRs.

(i) System for the treatment of liquid effluents

Where liquid metal coolants are used, the systems used in WCRs for the treatment of liquid effluents are not applied, and specific provisions need to be provided for each coolant technology. The situation may be similar to that in current WCRs only for spent fuel pools.

The systems used in WCRs for reactor coolant treatment and boron recycling are not used in HTGRs, because the reactor coolant is helium gas.

Before introducing spent fuel in the storage pool of some EIDs, the fuel needs to be cleaned. This is the case for SFRs where the fuel is cleaned by water, which needs to be treated and controlled as a liquid effluent. A similar process may also be defined for LFRs.

(j) System for the treatment of solid waste

There are some solid wastes that are specific to HTGRs, such as the replaceable graphite reflector in the prismatic HTGR and graphite dust collected during normal operation in the pebble bed HTGR.

Spent fuel is discussed in Section 3.6.1 regarding the areas of novelty in the management of waste and spent fuel.

(k) Equipment and floor drainage system

In accident conditions in current WCRs, the equipment and floor drainage system may have the capability to reinject highly contaminated liquids from the auxiliary buildings or secondary containment into the containment if the level of radioactivity in the effluent is too high to be treated in the short term (i.e. if storage would be needed before the treatment) or if the volume of fluids exceeds the waste treatment capacity. This capability deserves some analysis for EIDs, since the number of systems available to perform this function in accident conditions may be limited. In particular, this reinjection is not applicable to some EIDs, such as LFRs, SFRs and HTGRs.

For the sodium systems located outside the reactor vessel of an SFR, quick drainage of sodium may limit the consequences of a sodium leak. For example, draining the intermediate sodium loop (when combined with isolating the water system) mitigates the consequences of a steam generator tube rupture. Other design provisions that could be used for limiting the consequences of sodium leak include double walled pipes. The sodium loop used to remove decay heat by using atmospheric air is also equipped with a sodium draining system.

For HTGRs, where the primary coolant is helium gas, some safety functions that would be found in a WCR are not needed. Examples are monitoring the reactor coolant system for leaks during normal operation, and reinjection of highly contaminated liquids from the auxiliary buildings or secondary containment into the primary coolant circuit in accident conditions.

(l) Gaseous effluents

Gaseous effluents coming from the gas cover used by SFRs or LFRs are collected.

(m) Demineralized water reserve and associated system

As mentioned above, large PWRs avail on a demineralized water reserve to supply the auxiliary or emergency feedwater system for long term residual heat removal by the steam generators when the condensate storage tank has been emptied or is unavailable. This demineralized water reserve may not be applicable to non-water cooled EIDs (such as SFRs and LFRs). For HTGRs it is not needed and there are no safety functions similar to the function that is carried out by this system in PWRs.

(n) Fuel handling and storage

Some EIDs rely on fuel handling systems and storage that are significantly different from those used in WCRs. For example:

(i) Liquid fuel MSRs (except in the narrow liquid fuel concept) use a fuel handling and storage system that is adapted to liquid fuel. When fuel composition adjustment during operation is available, it relies on two way fuel transfer (i.e. from the reactor to the fuel system and from the fuel system to the reactor). This presupposes adequate facilities outside the reactor but within the NPP site. Provisions are necessary to maintain subcriticality in the liquid fuel that is out of the reactor and to remove the decay heat.
(ii) HTGRs pebble bed refuelling is performed on-line. The fuel pebbles are continuously loaded and discharged whilst the reactor is operating at power. Each fuel pebble loaded into the reactor circulates several times through both the reactor and the fuel handling system until its burnup reaches the design value. To enable this, the fuel handling system can identify and separate spent or damaged fuel pebbles. In prismatic HTGRs, batch refuelling of fuel blocks is performed during reactor shutdown using a block handling machine that is lowered through the vessel head.
(iii) Different cooling media for spent fuel may be used, such as air for HTGRs or sodium for SFRs.

(iv) In some cases, there may be dedicated design features for conditioning and transferring fuel from non-water coolants to a water pool.
(v) If fuel for non-water coolants is transferred to a water pool for long term storage, waiting periods can be required to reduce decay power levels so that safe coolability of the fuel assembly in water is ensured, depending on the specifics of the fuel assembly design.

(o) Other auxiliary systems

Some novel functions need to be achieved by auxiliary systems that are specific to non-WCRs. The following are examples of these functions and systems:

HTGR

(i) The helium purification system that removes the chemical impurities of the helium coolant. The graphite dust and radionuclides from the helium coolant are also removed by this system during normal operation in some designs.
(ii) The helium storage and supply system, whose main purpose is the control of coolant pressure within predetermined conditions during normal operation.

LFR and SFR

(i) Systems for storage, monitoring, and cleanup of the cover gas.
(ii) A coolant heating system and associated controls, used primarily to prevent coolant freezing — mostly when the reactor is shut down or starting up.
(iii) Systems for monitoring and control of the coolant composition, especially of oxygen.
(iv) LFRs and SFRs that have once-through steam generators have special requirements on water chemistry that are similar to those for WCRs.
(v) Provisions for detecting polonium in LFRs may need to be provided.
(vi) Sodium leak detection systems, since liquid sodium burns in air and reacts violently with water, and water systems may not be used in a compartment containing or likely to contain sodium. In addition, concrete needs to be protected in the areas where it could be in contact with sodium.

MSR

(i) MSRs with liquid fuel need a cover gas system for managing the salt volume variations, and for salt draining and filling operations.

3.2.8.4. Preliminary areas of novelty specific to transportable nuclear power plants

For a TNPP facility (marine or land based), different designs will have different approaches as to which auxiliary systems will be dedicated to the transportable portion of the facility and which to the stationary portion of the facility (onshore or permanently part of the site infrastructure). This is particularly important if the TNPP is designed to interface with a separate steam plant or other industrial facility. Due consideration of how a TNPP's systems interface with the site infrastructure is important to ensure that system reliability targets can be met with sufficient confidence.

3.2.9. Design against external hazards

3.2.9.1. General areas of novelty

From the point of view of design, external hazards can be divided into two broad categories: seismic hazard and all other hazards.

The seismic hazard, particularly the ground motion hazard, is related to the inertia forces generated in SSCs by the shaking ground. Therefore, the seismic action cannot be 'arrested' at the envelope of the buildings, and it reaches all plant components. The way it does so depends on the structural topology of the buildings (i.e. their geometry and the materials used) and their foundations. In addition, the way inertia forces affect the components depends on the components' physical configuration. Consequently, seismic design will follow the same rules and guidelines as in WCRs, if the civil structures (buildings) of the EIDs are similar and the SSCs in the two NPP types have similar configurations. However, some of the civil structures, and especially the SSCs, of some EIDs may be different to those of WCRs.

Design against all other hazards generally focuses on the envelope of the civil structures, since it is the barrier protecting the safety related components from the external hazard (e.g. wind, flood, accidental aircraft crash). Therefore, the design will follow the same rules and guidelines as in WCRs insofar as civil structures of the EIDs are similar to those of WCRs.

The structural topologies and building materials of several EIDs are basically the same as those currently used for WCRs.

Even if not very common, seismic base isolation systems have already been used in WCRs and guidance does exist in nuclear design codes.

Many novel designs show a large embedment of the containment and other safety structures, to the extent that those structures become semi-buried or completely buried in the ground. However, increasing the embedment of safety structures for the purpose of improving seismic performance, or for improving the resistance against external events, is a design option already contemplated by current nuclear design practices. A novelty that arises from this practice is that external flooding can be a significant concern, in particular — but not only — for non-water cooled EIDs such as SFRs.

Seismically induced sloshing in pool reactors is a technology dependent design problem that needs to be studied on a case by case basis. However, the current design practice already considers the sloshing in tanks and water pools, which is a problem of the same nature.

The seismic design of some technology specific components of EIDs might fall outside current practices and, hence, constitute a novelty. However, EIDs appear to have no significant implications in relation to design against external hazards.

3.2.9.2. Areas of novelty related to small size, multiple modules or modularity

No areas of novelty that are specific to SMRs were identified.

3.2.9.3. Areas of novelty specific to non-water cooled reactors

No areas of novelty that are specific to non-WCRs were identified.

3.2.9.4. Preliminary areas of novelty specific to transportable nuclear power plants

The design of TNPPs (marine or land based) against external hazards during navigation or transportation conditions follows completely different rules than traditional designs and are not covered by the current IAEA safety standards.

3.3. CONSTRUCTION

3.3.1. General areas of novelty

There are manufacturing and construction steps and sequences from the source (e.g. factory) to the deployment site, with their associated handovers between supply chain parties. The conditions during transport and deployment (such as temperature, humidity and potential for modules or components sitting in a non-ideal environment for months or longer) are also relevant for manufacturing and construction.

Advanced manufacturing methods, which may be adopted from other industries or developed specifically for nuclear applications, may include additive manufacturing, such as concrete 3-D printing, powder metallurgy with hot isostatic pressing, electron beam welding and diode laser cladding. In addition, novel materials and usage of recycled materials may be also envisaged.

There are also new techniques that may be used to manage construction and ongoing maintenance of fleets of the same or similar facilities; in particular, for procuring new modules or replacement components from manufacturers. Technologies such as configuration management information systems, building information modelling and digital twins are increasingly being tested and proposed before, during and after construction, and are likely to become standard practice for advanced designs.

As previously mentioned, some designs include structures and systems embedded or placed below grade and geotechnical construction. The use of subsurface structures introduces in-service condition monitoring challenges. Techniques and technologies (e.g. embedded instruments) may need to be introduced during manufacturing and construction to address post-construction and in-service inspections.

Water cooled and non-water cooled EIDs are often proposed to use combinations of inherent and passive safety features, which may reduce the number of safety related SSCs compared with existing facilities. The design process entails safety classification and code classification to be systematically used to demonstrate that the design of such systems will perform at a level of reliability that is in accordance with safety requirements. These processes are a standard part of applying a graded approach, and if appropriately used, will result in systems that match their safety significance. The outputs of these processes should interface with planning for manufacturing and construction through system and component specifications that reference appropriate proven practices.

EIDs may be deployed in remote locations, which may have less extensive or less reliable infrastructure than commonly found at existing NPPs. The use of remote operation, service and maintenance centres for fleet deployment of EIDs is part of the overall design for manufacturing and construction, considering conduct of operations and maintenance.

Regulators considering these technologies have highlighted that if manufacturing and construction processes are not implemented correctly for safety significant modules, they could result in potential latent issues [126]. Some regulators are implementing early engagement with proponents; for example, considering the evidence needed to demonstrate the safety outcomes of new manufacturing and construction methods and technologies. It may also help to plan the regulatory skills and research needed in anticipation of industry applications and licensee submittals.

3.3.2. Areas of novelty common related to small size, multiple modules or modularity

New approaches for manufacturing or construction, such as use of modular design and modular construction, and new manufacturing techniques are being adopted or developed across most parts of the nuclear power sector but are more likely to be proposed in newer reactor technologies.

Many EID developers are planning to rely on varying degrees of modularity. In particular, compared with conventional large WCRs, smaller sized EIDs and SMRs offer the ability to develop and implement novel manufacturing and construction methods, including increased factory fabrication and modularity. As these technologies become proven in practice, they may also be applied to larger NPPs or scaled up versions of the smaller facilities.

There will be site construction and establishment or improvement of infrastructure in advance of the deployment of the modular portions of the facilities (e.g. basemat, site infrastructure). This is expected to be integrated and reconciled in the planning for manufacturing and shipping of modules. Coordination between the factory and the site is very important so that, for example, when infrastructure is being created, there is detailed knowledge of the modules it is intended to serve. Assessment of tolerances and safe storage of constructed features until the modules can be shipped are examples of activities to be coordinated between the factory and the site.

There may be construction activities of new modules next to existing operational modules in the same facility. For example, there may be a 'construction island' within the facility, where the constructor may have jurisdictional rules in place for activities that may differ from, or conflict with, those of the 'operating island'.

3.3.3. Areas of novelty specific to non-water cooled reactors

The conditions under which non-WCRs are intended to operate will directly impact construction activities. For example, compared with currently operating WCRs, EIDs may have higher operating temperatures, harsher chemical environments and greater neutron fluence. Material selection, manufacturing practices, design for construction and planning for large scale maintenance[19] will need to reflect these operating conditions. For instance, concrete structures exposed to sustained elevated temperatures could experience spalling and

[19] Large scale maintenance includes major component replacements and facility refurbishment.

cracking, and this could require new concrete mixtures to be developed and evaluated to resist these ageing effects.

Lower reactor pressures may suggest that the traditional designs of a containment structure could be replaced. However, such structures will still need to be sufficiently leaktight to mitigate releases and provide protection from external events and accidental loadings.

3.3.4. Preliminary areas of novelty specific to transportable nuclear power plants

Transportable reactors (such as marine vessels and land based microreactors) may also require additional unique considerations. The use of reactors for civilian purposes[20] in marine environments introduces the possibility of applying marine construction standards to facility design. The use of alternative technical standards, such as marine architectural standards, may need to be demonstrated to be of equivalent effectiveness to existing nuclear practices.

If TNPPs are to be manufactured in a country different from the country of deployment, the design and manufacture of SSCs may consider the regulatory requirements in the country of deployment. Where a TNPP is designed to be relocated with fuel in the core or within the facility, transport design requirements and transport regulations are also to be integrated.

Some of the discussions in Section 3.3.1.1 on general areas of novelty are also relevant to TNPPs.

3.4. COMMISSIONING AND OPERATION

3.4.1. General areas of novelty

The relevant general areas of novelty are as follows:

(a) Extensive use of passive design features and systems

The operation of passive systems requires little or no external power or control from human operators or automated I&C systems (see Section 3.2.7). This has an impact on the methods used to routinely test the operability of these passive systems. In particular, the availability of passive safety features

[20] This Safety Report acknowledges that marine practice has longstanding pedigree in military nuclear marine applications, but access of civilians to operational experience is restricted.

or alternative safety arrangements may be different during periods when, for example, a reactor pressure vessel is open. Since the use of passive systems is extensive in EIDs and is relatively new in the nuclear industry, experience in the operation, maintenance, surveillance and periodic testing is limited. Additionally, as the performance of passive systems strongly depends on the physical boundary conditions (such as pressure and temperature) and the plant might not be easily configured to conditions where the passively delivered functions are needed, the representativeness of the commissioning tests and periodic tests could be difficult to establish.

(b) Combined activities (co-activities) related to the operation of EIDs [98]

The term ‘co-activity’ is used to describe a situation where multiple units at the same site are operated by a single team (including control room personnel, field operators and maintenance staff) from a single MCR. While some of the following topics are applicable to a few operating WCR designs, some EIDs might significantly expand this trend. Notable areas of novelty include the following:

(i) Limited or no staff present during normal operations.
(ii) Operation of multiple units from the same MCR. The handling of outage, incidental or accidental conditions in one or more units needs to be prevented from adversely affecting the safe operation of units in normal conditions.
(iii) Provision of one or more supplementary control rooms for units that require the intervention of large numbers of persons, to avoid disturbing the operation of units in normal condition. Arrangements for communication and coordination between the different rooms will also be needed.
(iv) Measures to ensure that for each operator action, whether from the control room or in the field, the action is performed on the correct unit.

(c) EIDs including co-activity plant systems

The design of multi-unit EIDs often has some plant systems shared by all or some units. This impacts I&C architectures, control rooms, control room systems and electrical power systems, and consequently operator training and plant familiarization.

(d) Optimization of maintenance

Maintenance may be affected by the longer operation cycles (as is the case for some EIDs) and reduced inventory of active components (e.g. valves, pumps).

(e) More extensive use of remote monitoring and signalling for testing, inspection, maintenance and control

Although remote support centres for large reactors already exist, they are intended mainly for use in emergency situations. Where the same EID is used for many units at different sites, some projects also consider remote centres for monitoring and aiding in more normal, everyday situations; for example, the use of remote control of reactor circuits and remote condition based maintenance monitoring techniques.

(f) Electrical grid integration

Smaller, less stable electricity grids (a possible consequence of remote siting) and increased use of variable renewable generators may require EIDs to operate flexibly. Designs may therefore need to be resilient to grid imposed short term shutdowns/restarts and have the ability to load follow.

(g) District heating systems

In view of the intended use of some EIDs to support district heating and other applications, designs may have different resilience needs to lengthy shutdown periods when heating is not required at all or only on a minimal level for providing hot water.

(h) Refuelling

Some designs have a long refuelling period with, in some instances, no on-site refuelling option and no local spent fuel storage. Where fuelling and refuelling take place on-site, the appropriate facilities will be required.

3.4.1.1. Areas of novelty related to small size, multiple modules or modularity

(a) Integrated designs

Some designs are based on primary circuits that are fully integrated within the reactor vessel (including steam generators, pumps, pressurizer and control rods). This has possible implications on maintenance; in-service inspection, surveillance and testing; radiation protection; and access to components.

(b) Modular construction

The use of modular construction techniques for EIDs implies that a significant percentage of plant construction is made by assembling subsystems (i.e. modules) built and verified at the factory and then transported to, and assembled at, the construction site. Thus, new inspections and verification may be needed for the assembly of the modules, the integrity of the modules (absence of damage and malicious tampering) during transportation and to confirm this integrity on-site.

(c) Staged construction and multi-unit construction

Some multi-unit EID projects consider staged plant construction, where additional units are constructed while the first ones are already in operation. Thus, it is necessary to ensure that construction or renovation works do not disturb the operating units to the point of causing safety or security issues.

Similarly, commissioning and operation activities may occur on adjacent modules that are in different operating states. This may introduce different types of initiating events to be considered in the plant safety case.

Common cause events could be triggered by equipment shared between modules (including portable equipment used for commissioning and maintenance).

Where commissioning factory tests have included the first reactor criticality and power tests, or other significant tests with nuclear safety implications, then additional testing is required at the site to verify that nothing has changed since these tests were carried out.

(d) Electricity grid connection

Some SMRs may be deployed in regions where there is no substantial electric grid infrastructure.

(e) Radiation protection

In terms of radiation protection programmes, there may be specific challenges for different EIDs. Many of these are unknown at this stage and need to be further explored. Potential issues related to occupational radiation protection that have already been identified are how maintenance operations and decommissioning, radon ingress and gamma radiation from naturally occurring radioactive material in construction materials will impact radiation doses to workers. In the case of public exposure, routine discharges are expected to be lower than with existing reactors, but this may depend on reactor design and operating conditions.

3.4.2. Areas of novelty specific to non-water cooled reactors

No areas of novelty specific to non-WCRs were identified.

3.4.3. Preliminary areas of novelty specific to transportable nuclear power plants

The following areas of novelty specific to TNPPs were identified.

(a) Commissioning and operation

To provide confidence that a floating reactor platform (ship) is adequately protected from external hazards, it will be necessary to examine possible failures of those parts of the floating platform that provide this protection; in particular, protection of the reactor vessel is of importance. Hence, it is necessary to identify and integrate into surveillance tests and emergency preparedness the verification and response to failure of those parts of the floating platform.

Testing and maintenance regimes include all nuclear and marine equipment that provides a nuclear safety function, whether directly or indirectly.

(b) Nuclear fuel

For some designs it is proposed that large modules containing new fuel may be transported to the site. Such an arrangement requires additional testing requirements at the site to ensure that no damage has occurred during transportation and that the components are fit for installation and service.

For some designs, refuelling occurs at a dedicated land based facility, and hence there are specific additional arrangements for fuel unloading and reloading.

3.5. NUCLEAR FUEL CYCLE

3.5.1. General areas of novelty

EIDs use a wide range of fuel types and fuel cycles. For WCRs, which may include PHWRs, these include conventional UO_2, which, however, may be enriched up to almost 20%. Some designs call for the use of MOX (U, Pu, Th) fuel with greater or smaller quantities of the principal components, depending on the design intention, such as utilizing surplus plutonium and creating new fissile material. The use of CERMET fuel (particles of ceramic fuel uniformly distributed in a metal matrix) is suggested for at least one PWR design. In the longer term, there is the possibility of using ATF, which is currently under investigation; this could utilize uranium nitride or silicide fuel and plated Zircaloy cladding.

The use of MOX fuel implies the need for spent fuel reprocessing. This could be the standard PUREX process for uranium–plutonium fuel. For thorium fuels, the THOREX process would be needed, which has not yet been developed to commercial scale.

Fuel types for other EID types are discussed in Sections 3.5.3.1–3.5.3.3.

3.5.2. Areas of novelty related to small size, multiple modules or modularity

There are no fuel or fuel cycle novelties in relation to small size, multi-module or modular NPPs.

3.5.3. Areas of novelty specific to non-water cooled reactors

3.5.3.1. High temperature gas cooled reactors

HTGRs fall into two groups — pebble bed and prismatic — and both deploy TRISO fuel in which small particles of (normally) UO_2 are coated in layers of graphite and silicon carbide. Proposed enrichments range from 8.5% to 20%. Alternatively, UO_2 may be replaced by PuO_2, uranium oxycarbide or thorium mixed with either uranium or plutonium. The coatings of silicon carbide and carbon make TRISO fuel resistant to mechanical damage and chemical attack, and no practicable means of reprocessing spent TRISO fuel has yet been devised. Since the presence of thorium in the fuel implies reprocessing, designs that propose this fuel are likely to be a long term prospect.

3.5.3.2. Molten salt reactors

Like HTGRs, MSRs also fall into two main groups: those in which the fuel is liquid and serves as heat source and coolant, and those in which the fuel is solid and the coolant is composed of molten salt. In the liquid fuel type, the fluoride salt of uranium, plutonium or thorium is dissolved in a carrier salt; alternatively, chloride salts may be used instead of fluoride. MSRs are very flexible with respect to fuel and may employ a thermal or a fast neutron spectrum. The use of liquid fuel allows volatile fission products to escape and be stored, and also gives the possibility for the fuel to be continuously processed during operation. Some designs include a breeder blanket, which may consist of uranium or thorium.

3.5.3.3. Fast reactors cooled by liquid metal

As previously explained, the liquid metal coolant of a fast reactor may be sodium (for SFRs), lead or a lead–bismuth eutectic alloy (for LFRs). In all cases, the fuel is often UO_2 or conventional MOX (U, Pu), although metallic fuel (U–Pu–Zr alloy), carbide and nitride fuels (U or Pu) are sometimes considered. The use of a fast spectrum necessitates higher enrichments, and HALEU fuel enriched up to 20% ^{235}U is normal. There is little manufacturing experience for some of the non-conventional fuels.

3.5.4. Areas of novelty specific to transportable nuclear power plants

TNPPs can use any of a wide range of EIDs, all of which have been briefly described in Sections 3.5.2 and 3.5.3. TNPPs may use ‘lifetime cores’, from which fuel is removed only when the reactor is dismantled. This usually requires high ^{235}U enrichment levels.

3.6. MANAGEMENT OF RADIOACTIVE WASTE AND SPENT FUEL

3.6.1. General areas of novelty

The wide range of EIDs implies that the waste and spent fuel generated could be diverse, ranging from low level waste to high level waste and including spent fuel of different types, burnups and thermal powers, waste from various different types of reprocessing and various novel waste types (see, for example, Section 3.6.3). Every national waste management system would need to be able to accommodate the diversity of waste and spent fuel arising in the country. Suitable plans and procedures for the processing of the individual waste and spent fuel types would need to be implemented in every country to make the waste suitable for disposal. Planning for national waste management systems would need to take account of the diverse range of waste streams present in the country and the interdependencies between the steps in the overall waste management process, from processing to disposal or release from regulatory control.

The widespread adoption of novel reactors within a country could increase the geographical spread of on-site radioactive waste and spent fuel storage. This would tend to go against the preference in the safety standards for centralized storage of waste and would have implications for the maintenance of safety and security in the long term.

3.6.2. Areas of novelty related to small size, multiple modules or modularity

Some factory fuelled reactors might be treated as waste packages. Entire reactors, which some developers of SMRs have suggested could be disposed of in one piece, would constitute novel radioactive waste.

3.6.3. Areas of novelty specific to non-water cooled reactors

Various other novel radioactive wastes could potentially be generated from non-water cooled EIDs, including the following:

(a) Graphite from HTGRs, which is intimately mixed with SNF and, for prismatic type HTGRs, used as a moderator.
(b) Halide salts containing nuclear fuel in solution from MSRs. Radioactive off-gases are also produced from MSRs in relatively greater quantity than from other reactor types.
(c) Sodium contaminated components (e.g. SNF) from the reactor or the secondary circuit(s) of SFRs, which would complicate waste handling.
(d) Lead and lead–bismuth coolants from LFRs, which are both radioactive and toxic.

If these innovative technologies are widely adopted, these waste types could be produced in large volumes. It is likely that further research would be needed to investigate certain processes (e.g. the release of halide gases from solidified MSR salt fuels and implications such as the combination of fluorine gas with uranium to form UF_6 and mobilize fissile material) and determine the optimal plans for the management of novel waste types from EIDs (e.g. processing of metallic Zr–U or Zr–U–Pu fuels).

3.6.4. Areas of novelty specific to transportable nuclear power plants

As previously mentioned, TNPPs are primarily designed to operate in remote locations where there is no grid connection. Some reactor designs broadly follow the PWRs already used in nuclear powered vessels, while others opt for fast reactors. In all cases, the intention is that the reactors should be capable of operating without intervention for a period of years, with fuel being charged and discharged at a separate shore based facility that has the specialized equipment required to perform such operations safely. With respect to spent fuel management, therefore, there is little to differentiate TNPPs from other EIDs.

Descriptions of TNPPs (see, for example, Ref. [2]) are generally silent on the matter of the operational wastes that would primarily arise from coolant cleanup. While it is possible that a reactor could be designed to operate without cleanup, this would limit personnel access to certain areas or require additional shielding. For this reason, it seems likely that coolant cleanup will be performed so that wastes such as ion exchange resins will be generated. The subsequent storage of these wastes will need to recognize the hazards that arise on a seagoing vessel; the same applies to the reactor itself.

3.7. DECOMMISSIONING

3.7.1. General areas of novelty

The introduction of nuclear cogeneration, such as hydrogen production and use of nuclear heat for industrial use, will impact decommissioning by placing additional hazards in the vicinity of the nuclear power plant.

3.7.2. Areas of novelty related to small size, multiple modules or modularity

The use of TNPPs, integrated reactor designs and one-time fuelling is likely to encourage the use of centralized decommissioning facilities (as opposed to on-site decommissioning). The ARIS database [4], for example, includes several designs of reactors that have to be sent to a centralized facility for fuel removal, dismantling and waste management. In some cases, this could require transport to another country. Such use of a centralized facility for decommissioning can reduce significantly the time necessary for on-site decommissioning and for cleanup of the site. Reuse of the site could be possible soon after the removal of the reactor, and a replacement of the old reactor with a new one could be feasible. There could be specific interfaces between the decommissioning of the previous reactor and the site evaluation for the replacement reactor that need to be considered, and some specific requirements may apply.

Decommissioning of SMRs located in remote regions may require environmental cleanup to be performed under extreme climatic conditions.

3.7.3. Areas of novelty specific to non-water cooled reactors

Contamination of components by liquid metal coolant and liquid fuel is a major departure from water cooled technology. Relatively few Member States have recent experience of working with these materials and, in general, these types of fuel are not considered by the current IAEA safety standards.

3.7.4. Areas of novelty specific to transportable nuclear power plants

Section 3.7.2 includes a discussion of TNPPs.

3.8. EMERGENCY PREPAREDNESS AND RESPONSE

3.8.1. General areas of novelty

Several areas of novelty, already discussed in relation to the design, commissioning and operation of the EIDs, may have implications for the on-site and off-site emergency arrangements. They are as follows:

(a) The use of novel materials, as well as the proximity to industrial facilities (such as refineries, chemical or hydrogen production facilities) that might be co-located with some EIDs, might exacerbate non-radiation related hazards (chemical, explosions, fires) that would need to be considered in the overall emergency arrangements.
(b) New or reduced source terms associated with some EIDs (compared with existing large NPPs) have the potential to reduce the off-site consequences of a severe emergency and thereby decrease the size of emergency planning zones and distances and associated level of planning for off-site emergency response.
(c) Some EIDs are claimed to be designed to be self-sufficient and, thus, can have the capability to be located in remote areas. Such a location would increase the response times needed by external (off-site) emergency services (e.g. firefighting, medical teams) to mount an effective response and, in general, would impact the associated emergency arrangements.
(d) There are claims that for some EIDs, the slower accident kinetics leading to a release of radioactive material into the environment (compared with WCRs) might allow for less urgency in the implementation of off-site public protective actions.

3.8.2. Areas of novelty related to small size, multiple modules or modularity

Co-location of EIDs with existing NPPs and multi-module plants could result in an event simultaneously affecting several modules and units at the same site. This might significantly strain the resources (technical as well as human) necessary to effectively mitigate the on-site and off-site consequences of the emergency. This may also impose certain limitations on the use of dedicated emergency response facilities.

Some SMRs might rely on remote services or remote centres (e.g. those providing necessary emergency services on-site) more extensively than current NPP technologies and, thus, impose challenges to the overall emergency arrangements needed for SMRs.

3.8.3. Areas of novelty specific to non-water cooled reactors

Use of novel materials (e.g. liquid metal coolants, molten salts) might pose new or different non-radiation related hazards that might jeopardize the effectiveness of the emergency response and might drive the emergency response goals differently. This will impact the design of the necessary emergency arrangements for these reactors.

3.8.4. Preliminary areas of novelty specific to transportable nuclear power plants

TNPPs may require emergency arrangements that differ from those of stationary reactors. This is particularly relevant for the arrangements needed to respond effectively along the transport route of the components loaded with fuel or containing activated primary coolant or effluent. These arrangements have to cover the route from a factory to the deployment site and, if applicable, to a service and refuelling facility, in an integrated manner.

3.9. DEPLOYMENT MODELS

3.9.1. Areas of novelty related to small size, multiple modules or modularity

The strategy for the deployment of SMRs relies on the idea of factory fabricating a series of SMRs (or parts of them) and potentially selling them as ‘off the shelf’ items, possibly even before a construction licence or operating licence has been issued. This approach differs from that for conventional WCRs, where the component manufacturing and construction stages normally follow regulatory approval.

The different options for the deployment of SMRs (series fabrication and 'turnkey'[21] contract) imply that the vendor/manufacturer and the purchaser/operator of an SMR could be located in different countries. If so, arrangements are necessary to give confidence to the regulatory body of the purchaser country that the design and manufacture of the units comply with the regulations in force in the purchaser country. While this is not dissimilar from current arrangements when nuclear technology is exported, the off-site manufacture of a complete reactor adds a layer of complexity.

3.9.2. Preliminary areas of novelty specific to transportable nuclear power plants

The deployment models for a TNPP can be very different from those of conventional WCRs. It is useful to consider two possible scenarios, both of which apply to either land based or marine based TNPPs:

(a) Scenario 1: Transport of the TNPP to the site of installation with the reactor empty of fuel. New fuel is then loaded at the site; at the end of life, it is unloaded at the site and either stored there or sent to another facility. The TNPP, now empty of fuel but still containing radioactive material, may be transported to another site or back to the manufacturer.
(b) Scenario 2: Transport of the TNPP to the site of installation with fresh nuclear fuel already present in the reactor. After a period of operation, the TNPP may be taken, complete with irradiated fuel, to another site or sent back to the manufacturer.

Scenario 1 does not imply any significant difference in terms of transport from the practices used for conventional, stationary NPPs. Scenario 2 creates several new issues for transport because it is a completely new practice and, so far, the concept of transporting a reactor with its fresh nuclear fuel or SNF is not specifically covered in existing safety standards or specific safety provisions for the transport of radioactive material.

[21] In a turnkey project, a single contractor or a consortium of contractors takes overall technical responsibility for the entire project.

4. MAPPING OF APPLICATION OF SAFETY STANDARDS

4.1. INTRODUCTION

The high level review described in Sections 2 and 3 aimed to provide insights to be used as guidance for a further and deeper assessment of the applicability of the IAEA safety standards to EIDs. This section is based on a screening of these standards focusing more on their safety objectives and less on the actual vocabulary used, given that this might sometimes be oriented to WCR technology. In Section 3, areas of novelty arising from EIDs were identified and divided into four groups: (1) general areas of novelty; (2) areas of novelty related to small size, multiple modules or modularity; (3) areas of novelty specific to non-WCRs; and (4) preliminary areas of novelty specific to TNPPs.

This section compares the areas of novelty with the requirements and guidance in the IAEA safety standards[22]. Inferences are then drawn regarding the comprehensiveness of the standards and their applicability to EIDs, classifying these inferences into three groups (whilst acknowledging that there will be some overlap between them):

(1) Areas where the standards are applicable to EIDs.
(2) Areas where the standards are non-applicable to EIDs. 'Non-applicable' means that additional guidance is needed to put the existing standard into context by, for example, indicating that a particular recommendation is specific to WCRs or by including examples that are specific to EIDs.
(3) Identified gaps and areas for additional consideration. Parts of the safety standards that may not fully cover the areas of novelty of EIDs are named 'gaps' herein. In some cases, although gaps were not identified, the review outlined additional considerations that may be necessary to complement the information in the IAEA safety standards.

Again, the classification of requirements and guidance presented above is the result of a high level review, and in some cases an area that may not be applicable is also related to a gap and vice versa. These categories give a general indication and are complementary.

[22] For MSRs, the safety standards review was done only for SSR-2/1 (Rev. 1) [3]. For TNPPs, the results must be regarded as preliminary.

This section, and this Safety Report in general, does not aim to assess the significance (for safety or otherwise) of a particular item of non-applicability or gap; this will probably be addressed in future work.

4.2. SITING

4.2.1. IAEA Safety Standards Series No. SSR-1, Site Evaluation for Nuclear Installations

4.2.1.1. Areas of applicability

Safety requirements for site characterization and site safety evaluation for nuclear installations are established in SSR-1 [18]. No area of novelty was identified in Section 3 that is not already considered in the siting of the currently operating large WCRs. Therefore, the safety requirements in SSR-1 are considered to be applicable to all land based nuclear installations, including EIDs.

A floating TNPP could be considered as land based if it is fixed at a particular geographical position. In that case, SSR-1 requirements for characterization and safety evaluation of the site would apply, with little difference with respect to regular, ground supported, WCRs.

4.2.1.2. Areas of non-applicability

SSR-1 [18] was written for stationary, land based reactors. Although it would be largely applicable to reactors that are moored at a location, this may not be the case for marine based and transportable reactors during navigation or transportation (i.e. when there is no site).

4.2.1.3. Identified gaps and areas for additional consideration

No gaps were identified.

4.2.2. IAEA Safety Standards Series No. SSG-35, Site Survey and Site Selection for Nuclear Installations

4.2.2.1. Areas of applicability

The scope of SSG-35 [25] is not limited to any particular reactor technology or type of nuclear installation.

4.2.2.2. Areas of non-applicability

SSG-35 [25] is not applicable when a (stationary) site does not exist, but it is applicable to the selection of sites to place marine based NPPs or TNPPs.

4.2.2.3. Identified gaps and areas for additional consideration

TNPPs are not covered unless placed at a specific site.

4.2.3. IAEA Safety Standards Series No. SSG-79, Hazards Associated with Human Induced External Events in Site Evaluation for Nuclear Installations

4.2.3.1. Areas of applicability

No fundamental areas of novelty specific to EIDs are relevant to the topics covered by SSG-79 [30]. It follows that this guide is applicable to EIDs.

4.2.3.2. Areas of non-applicability

SSG-79 [30] is not applicable when a (stationary) site does not exist.

4.2.3.3. Identified gaps and areas for additional consideration

TNPPs are not covered by SSG-79 [30] if they are not placed at a specific site.

4.2.4. IAEA Safety Standards Series No. NS-G-3.2, Dispersion of Radioactive Material and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants

4.2.4.1. Areas of applicability

The scope of NS-G-3.2 [31] is not limited to any particular reactor technology, since radioactive sources are considered to be input data for the user of NS-G-3.2. It follows that this guide is applicable to EIDs.

4.2.4.2. Areas of non-applicability

NS-G-3.2 [31] is not applicable when a (stationary) site does not exist.

4.2.4.3. *Identified gaps and areas for additional consideration*

TNPPs are not covered by NS-G-3.2 [31] if they are not placed at a specific site.

4.2.5. IAEA Safety Standards Series No. NS-G-3.6, Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants

4.2.5.1. *Areas of applicability*

The scope of NS-G-3.6 [29] is not limited to any particular reactor technology. It follows that this guide is applicable to EIDs.

4.2.5.2. *Areas of non-applicability*

NS-G-3.6 [29] is not applicable when a stationary site does not exist. However, some of the geotechnical aspects covered by NS-G-3.6 will need to be investigated for sites where floating or transportable reactors are docked.

4.2.5.3. *Identified gaps and areas for additional consideration*

TNPPs are not covered by NS-G-3.6 [29] if they are not placed at a specific site.

4.2.6. IAEA Safety Standards Series No. SSG-9 (Rev. 1), Seismic Hazards in Site Evaluation for Nuclear Installations

4.2.6.1. *Areas of applicability*

The scope of SSG-9 (Rev. 1) [26] is not limited to any particular reactor technology. Consequently, this guide is applicable to EIDs.

4.2.6.2. *Areas of non-applicability*

SSG-9 (Rev. 1) [26] addresses land based stationary NPPs. In other words, it appears to exclude floating TNPPs but could still be relevant when such a plant is permanently moored, or even temporarily anchored, close to land. Hence, SSG-9 is not applicable when a (stationary) site does not exist.

4.2.6.3. Identified gaps and areas for additional consideration

The following gaps are identified:

(a) Transportation related loads for TNPPs will very likely exceed the seismic loads in most specific sites;
(b) TNPPs are not covered by SSG-9 (Rev. 1) [26] unless they are placed at a specific site.

4.2.7. IAEA Safety Standards Series No. SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations

4.2.7.1. Areas of applicability

The scope of SSG-18 [27] is not limited to any particular reactor technology. It follows that this guide is generally applicable to EIDs.

4.2.7.2. Areas of non-applicability

SSG-18 [27] addresses land based stationary NPPs. In other words, it appears to exclude floating TNPPs but could still be relevant when such a plant is permanently moored, or even temporarily anchored, close to land. Hence, SSG-18 is not applicable when a (stationary) site does not exist.

4.2.7.3. Identified gaps and areas for additional consideration

TNPPs are not covered by SSG-18 [27] if they are not placed at a specific site.

4.2.8. IAEA Safety Standards Series No. SSG-21, Volcanic Hazards in Site Evaluation for Nuclear Installations

4.2.8.1. Areas of applicability

The scope of SSG-21 [28] is not limited to any particular reactor technology. It follows that this guide is applicable to EIDs.

4.2.8.2. Areas of non-applicability

SSG-21 [28] addresses land based stationary NPPs. In other words, it appears to exclude floating TNPPs, but could still be relevant when such a plant is permanently moored, or even temporarily anchored, close to land. Hence, SSG-21 is not applicable when a (stationary) site does not exist.

4.2.8.3. Identified gaps and areas for additional consideration

TNPPs are not covered by SSG-21 [28] if they are not placed at a specific site.

4.3. DESIGN

4.3.1. IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design

4.3.1.1. Areas of applicability

The safety requirements related to the management of safety in design (Requirements 1–3) and several principal technical requirements (Requirements 4–6 and 8–11), general plant design requirements (Requirements 14, 15, 18, 22–29, 32–34 and 36–42) and specific plant system requirements (Requirements 51–53, 69, 70 and 75) in SSR-2/1 (Rev. 1) [3] are generally applicable to EIDs.

Requirements 43–46, which are related to the reactor core and associated features, are fully applicable to water cooled EIDs. Considerations on non-water cooled EIDs are presented in Sections 4.3.1.2 and 4.3.1.3.

4.3.1.2. Areas of non-applicability

The following areas of possible non-applicability were identified:

(a) Significant fuel damage and core damage, including core melting scenarios, are claimed to be precluded in some EIDs, such as HTGRs and in some MSR designs, so references to core melting and severe accident in various requirements may not be directly applicable. For example, para. 4.13A of Requirement 7, para. 5.1 of Requirement 13, para. 5.30 of Requirement 20, para. 5.27 of Requirement 20, Requirement 42 and para. 6.44B of Requirement 68.
(b) Requirements 43–46 on the reactor core contain some terminology and examples that are predicated on water cooled technology experience. These either may not have the same paradigm as that in some EID technologies or

may not be applicable at all in others (additional discussion is presented in Section 4.3.1.3).

(i) Regarding Requirement 43:
  — Some designs (such as MSRs and HTGRs) may not use fuel elements and assemblies as traditionally understood, and therefore structural integrity requirements would need to be interpreted to, for example, the equivalent physical barriers.
  — This requirement does not appear to be expressed so as to enable users to translate and justify its meaning with respect to some non-WCR technologies, which suggests that the fundamental safety objective expressed in the requirement may not be sufficiently clear to allow interpretations and justification. For example:
    - Paragraph 6.1: (i) Buildup of helium does not occur in all types of reactor, so the paragraph may not apply to HTGRs, and (ii) the process of deterioration associated with static and dynamic loading, including flow induced vibrations and mechanical vibrations, may not be manifest in all types of non-water cooled EID.
    - Paragraph 6.2: For liquid fuel MSR concepts with liquid fuel in tanks and pipes, normal operation should preclude leaks. Therefore, it may be appropriate to classify leaks from pipes and tanks as design basis accidents to be prevented. Hence, this paragraph may require interpretation to support this approach.
    - Paragraph 6.3: Fuel handling might not be conducted on-site for all types of reactor. For example, some reactors are designed to not require refuelling or fuel reshuffling during their whole lifetime (30 years).

(ii) Regarding Requirements 45 and 46 (para. 6.11), some EIDs will be fuelled for the reactor lifetime and will not require refuelling at the reactor facility site. In addition, for Requirement 45, para. 6.5, the notion of wear out may not apply to control devices used by some EIDs, such as liquid fuel MSRs without control rods, and some fast neutron SMRs in which reflectors are used.

(iii) Regarding Requirement 46, para. 6.8, failure of a control rod to insert is provided as an example but may not be applicable to EIDs with other types of control device.

(c) Requirement 50, para. 6.17 is not applicable to MSRs with liquid fuel circulating in the reactor. In the case of leakage of the reactor vessel or the fuel circuit, the reactor is shut down. It is applicable to MSRs with solid fuel and MSRs with liquid fuel separated from the coolant (i.e. liquid fuel in tubes or pins).

(d) Requirements 54–58 on the containment structure and containment system are largely written in a technology neutral manner and, thus, are mostly applicable to EIDs. However, Requirements 55–58 are related to the confinement of radioactive substances by the containment structure. In some non-WCR EIDs, such as HTGRs, this function may be performed by the TRISO fuel. If so, these requirements are not applicable to the reactor building, which remains necessary to ensure the other functions associated to the containment system (protection against external events and radiation shielding).

The following considerations are relevant to the applicability of these requirements to these EIDs (additional comments are presented in Section 4.3.1.3):

(i) The containment is the ultimate barrier to confine radioactive material in order to prevent unacceptable releases to the atmosphere in the case of accident conditions in current WCRs. On the other hand, some EIDs, especially HTGRs, apply a confinement concept rather than a containment system, on the assumption that sufficient retention is ensured by other barriers. Currently, most HTGR designs are not equipped with a containment structure. In general, the containment structure (if there is one) may be considered to have minor importance for retention of radioactive substances in these EIDs.

(ii) Requirement 56 and para. 6.21 address the isolation of the containment structure to ensure the confinement of radioactive substances. This concerns the designs where there is a containment structure (as in WCRs) that is equipped with penetrations. For some EIDs, such as HTGRs, where such structure may not be used, Requirement 56 may not be relevant.

(iii) Requirement 56 and para. 6.22 refer to the coolant pressure boundary. Some EIDs are operated at low pressure, so the term ‘coolant pressure boundary’ might not be applicable.

(iv) Paragraph 6.27 addresses flow routes between separate containment compartments and openings between compartments. This recommendation may not be applicable to very small containment designs.

(v) The redundancy requirement addressed in para. 6.28 may not be applicable to passively air cooled containments.

(vi) Paragraph 6.28 requires measures to reduce the pressure inside the containment. However, in the case of some EIDs, such as LFRs, no pressure increase is expected. For SFRs, sodium leakage and combustion may be a cause of pressure increase. For these EIDs, no systems to cope with accidental release of high energy fluids are required. The requirement is not relevant for HTGRs with TRISO fuel.

(vii) Paragraph 6.29(b) addresses the highly energetic phenomena (deflagration or detonation) that can jeopardize the structural integrity and the leaktightness of the containment structure, as well as the operation of the safety systems inside this structure. This requirement is relevant for the reactor building of EIDs, taking into account the hazardous substances that could be released in these reactors (e.g. hydrogen resulting from sodium-water reaction in SFRs). Nevertheless, for some EIDs, such as HTGRs, the leaktightness of the structure may not be an issue.

4.3.1.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) Throughout SSR-2/1 (Rev. 1) [3], requirements are formulated as applying to the plant or the unit, which is applicable to EIDs as written. For modular reactors, additional explanatory guidance might help with the interpretation and application also on a module level, considering the complementarity of module, plant and site level views.

The formulation of Requirements 12, 17, 19, 20, 30, 31, 33 and 35 for plant design is generally applicable, but the following aspects are not considered:

(b) Requirement 12: Non-WCRs may raise specific safety issues in the treatment and storage of wastes (radioactive and chemical wastes). For example, this is the case for wastes issued from the sodium purification system of SFRs that may combine a high content of tritium and sodium.
(c) Requirement 17, para. 5.16: Even though the list of hazards indicated in this requirement does not intend to be comprehensive, examples of internal hazards that are specific to EIDs are not explicitly mentioned (e.g. sodium fire and sodium–water reaction for SFRs, graphite fire for HTGRs).
(d) Requirement 17, para. 5.21A: Identification of the items ultimately necessary to prevent an early radioactive release or a large radioactive release is a prerequisite for a proper implementation of this requirement in practice. This identification is particularly relevant for EIDs, for which less regulatory and operating experience is available. The identification of these items and the definition of the associated safety requirements are also needed for the EIDs designed using the functional containment concept. For example, the ultimate provision for limiting radiological releases from HTGRs is provided by the TRISO fuel, combined with the design

characteristics of the plant, which allow an inherent limitation of the loads on the TRISO fuel during any accident conditions.

(e) Requirements 13 and 19: While both requirements are applicable, their implementation for EIDs will rely on interpretations, using explanatory guidance where it exists, of terms such as DBA, DEC and severe accident in relation to EIDs. Examples on how these fundamental concepts apply to the different types of EID would be beneficial, given the current lack of experience with these designs. For microreactors, for example, adequate and proportionate definitions of operational states, DBAs, DECs and severe accidents is likely quite different from those for large WCRs, and will have substantial impact on the interpretation of numerous other requirements.

(f) Requirement 20: The requirement is largely applicable as written. The interpretation of fundamental concepts such as containment, barrier or severe accident for some EIDs might be assisted by explanatory guidance, as for Requirement 19, and by the IAEA Safety Glossary [127]. This observation applies, for example, to MSR concepts where radioactive materials (i.e. liquid fuel, fission gases) might be located in different areas during normal operation (the liquid fuel may be in the core vessel, in drain tanks or in the molten salt processing system). The notions of barriers and DiD are applicable to each location having radioactive materials, but they may not be explicitly covered. In para. 5.30, consideration of core melting scenarios is explicitly required, which might not be applicable to, or proportionate for, some EIDs (and their fuel concepts and/or power levels and (volumetric) power density). Paragraph 5.30 might be interpreted as relating to a suitable severe accident scenario, but a technology neutral definition, which could be supported by examples for various EIDs in explanatory guidance, is not available. In general, significant fuel damage and core damage, including core melting scenarios, are claimed to be precluded in some EIDs, such as HTGRs and some MSR designs. A technology neutral approach independent from core melting would be useful for applying the logic behind the affected requirements and associated Safety Guides, but it is not available.

(g) Requirements 24 and 25: Further consideration for passive systems regarding the applicability of the single failure criterion and CCFs may be useful.

(h) Requirement 30: Parts related to the qualification of items important to safety may need adaptation for those EIDs for which safety depends on a combination of several plant properties. For example, the efficiency of the conduction cooldown of HTGRs includes considerations such as adequate physical properties of graphite for the core design, core power and radiative property of the reactor vessel throughout its lifetime.

(i) Requirement 31: Even though the list of mechanisms indicated in this requirement is not intended to be comprehensive, novel mechanisms of

EIDs that significantly influence the ageing of plant equipment are not explicitly mentioned (e.g. high operating temperature, chemical properties of the coolant).

(j) Requirement 33: The specific wording requires that: “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” [3]. If ‘unit’ is interpreted as ‘module’, a common external pool, confinement system or other shared safety features foreseen in some EIDs, these EIDs would not comply with this requirement. As an example, in the case of an MSR with multiple modules or units, several units may have a common molten salt processing system, fission gas recovery system or both. Safety provisions of existing WCRs might be shared between multiple units; for example, the multi-unit containments of CANDU (Canada Deuterium Uranium) reactors.

(k) Requirement 35: This requirement could be extended to other uses of NPPs. In addition, hazards introduced by the facility associated with an NPP could also be addressed as internal hazards in Requirement 17. Provisions to reject facilities causing unacceptably high risks for the NPP are not included.

Requirements 7, 16, 21 and 42[23] constitute a sound and applicable basis for the design of EIDs. However, the intent of these requirements may not be expressed in a technology neutral manner, and examples on how they could be applied to each type of technology are not included. The following considerations are also relevant:

(l) Requirement 7: This requirement is written in a technology neutral manner and is applicable to WCRs and EIDs, but its implementation in practice may be different for some EIDs, given that their approaches to DiD will differ. For example, regarding paras 4.11 and 4.12, a physical barrier might be achieved by TRISO particles for HTGRs and solid fuel MSR concepts. In addition, fission product retention in molten salt (in liquid fuel MSRs and solid fuel MSRs) may provide an inherent barrier to release for the MSR designs that naturally release fission gases from the salt during normal operation. Similarly, while para. 4.13A is ‘applicable as is’, specific guidance on its interpretation for EIDs, which utilize passive features on multiple levels of DiD, might be considered in explanatory guides.

(m) Requirement 16: While this requirement is applicable as written, specific guidance on PIEs for various EIDs could be considered in underlying guides.

---

[23] Requirements 13, 16, 19, 20 and 42 are closely linked and should be considered together.

(n) Requirement 42: The standard approach for advanced WCRs is based on the analysis of safety systems associated with the different levels of the DiD, notably the containment structure. For some EIDs, such as those using TRISO fuel particles, the safety approach requires a range of plant properties to act in concert ('holistically') to provide the confinement function. However, this requirement does not include these considerations.

Requirements 43–82 do not consider all the specific characteristics of the reactor technologies and the areas of novelty implemented in the design of EIDs.

Requirements 43–46 for fuel and core safety fully apply to light water SMRs and fundamentally to non-WCRs. However, given the significant differences between the cores of non-WCRs, additional considerations are important for the design. Some statements do not clearly represent the design requirements that would best apply to non-WCRs. In some cases, passive and inherent safety features in the design and operation of the new technologies are absent. In other words, the requirements established for WCRs may be either too prescriptive or too vague to accommodate the claimed safety advantages and improvements that are designed into, or are inherent in, EIDs.

(o) Requirement 43: Generally, the conditions and process of deterioration outlined do not depict the new issues associated with non-water cooled EIDs, such as higher temperatures compared with WCR conditions. Another example is that the liquid fuel of MSRs may be degraded by new phenomena that do not occur in WCRs, such as those leading to a modification of the solubility of the fissile materials.
(p) Requirement 43 (cont.): Some EIDs, such as HTGRs, may rely on the properties of TRISO fuel to retain fission products in both normal and accident conditions. Acceptance of such concepts will require demonstration that the claims made for the fuel's performance are valid and that the manufacturing processes are capable of consistently producing fuel of the required quality. Nevertheless, owing to the very high number of TRISO fuel particles used in the HTGR (and some MSR) cores, there will, statistically, be some TRISO particles that fail in normal operation and under accident conditions. The maximum ratio of TRISO fuel failures must be demonstrated by testing and ensured by adequate quality control during fuel manufacturing. The reactor must then be designed to cope with this ratio.
(q) Requirement 43 (cont.): The consideration of stresses associated with fuel handling does not account for fuel handling occurring off-site or by novel processes (draining of the liquid fuel in the case of some MSR designs).
(r) Requirement 44: The core geometry of WCRs must be maintained in all conditions (other than severe accidents) to allow the core to be cooled

efficiently and the control rods to be inserted. For liquid fuel MSRs, the salt can be cooled in any geometry if it is located in an area equipped with cooling systems or allowing natural cooling. Moreover, the shutdown of the reactor may be done by means other than control rods (e.g. draining the liquid fuel in tanks designed to maintain the fuel subcritical).

(s) Requirement 44 (cont.): Cases where control rods are not used or where other, equally viable, technological options can be demonstrated (e.g. certain fast neutron SMR designs using reflectors for control, some liquid fuel MSR designs) are not considered. In addition, the requirement does not fully cover the case of some EIDs, such as HTGRs, for which maintaining core geometry is not considered necessary for ensuring adequate core cooling.

(t) Requirement 44 (cont.): For liquid fuel MSRs, this requirement, which is related to the core geometry, could be substituted by a requirement related to the control of the fissile material concentration, by design and control of the physicochemical properties of the salt.

(u) Requirement 44 (cont.): In some other non-water cooled EIDs, shutdown can be achieved by other means than control rods (e.g. by using reflector withdrawal in certain fast neutron EIDs).

(v) Requirement 45: For fast neutron EIDs, reactivity variations may result from minor core geometry modifications (e.g. due to thermal expansion or contraction). This effect is more important than in WCRs. It may be managed by specific design provisions.

(w) Requirement 45 (cont.): In general, a reactor design may rely on all reactivity feedbacks to mitigate potential accident conditions, but some important feedbacks are not mentioned. For example, neutronic stability in some MSR designs may be achieved by inherent negative reactivity feedback caused by a decrease in salt density with temperature.

(x) Requirement 45 (cont.): For liquid fuel MSR concepts, the core stability may be affected by the delayed neutron fraction, which operators can modify by changing the fuel salt flow. This effect is not present in solid fuel MSRs and is not mentioned in the standard.

(y) Requirement 45 (cont.): The loss of reactivity control by accidental concentration of fissile material in the fuel salt of liquid fuel MSRs is not mentioned in the standard.

(z) Requirement 46: Some developers of non-water cooled EIDs claim that the strong negative temperature coefficient could substitute a shutdown system. For example, in some MSR concepts, reactor shutdown can be achieved by inherent negative reactivity feedback effects. Thus, the reactor is shut down by a certain temperature increase compared with the normal fuel temperature. These possibilities are not covered in this requirement.

Requirements 47–53 concern the RCS. The identified gaps are as follows:

(aa) Requirements 47 and 48 do not include components (e.g. tank, pipe) that are likely to contain fuel and fission products generating decay heat. In particular, systems containing fission gases, used in MSR concepts with extraction of these gases from the core, are not considered.
(ab) Requirement 49 does not address the possibility that the content of fissile material and fission products in the fuel salt could modify the salt's chemical properties, leading to safety concerns in the liquid fuel MSR concepts; for example, extensive corrosion of the structure, deposition of metallic elements and concentration of fissile materials.
(ac) Requirement 50 regarding coolant cleanup is generally applicable to EIDs. However, specific issues associated with control of the chemical behaviour of a molten salt, and specifically with corrosion control, are not covered. In the same way, LFRs require adequate corrosion control from a point of view of core coolability; this is also not mentioned.
(ad) Requirement 51 does not consider the removal of residual heat from any locations in MSRs containing fuel, fission gases or both.
(ae) Requirement 52 does not account for possible overcooling of an MSR and its fuel, which could (i) lead to unacceptable freezing of the molten salt, and (ii) modify the physicochemical properties of the salt, which could entail the risk of fissile material concentration (e.g. by precipitation). For liquid fuel MSRs, emergency cooling is needed for all locations where fuel could be present and at any possible location of the fission gases (in case of extraction of the fission gases from the core). Owing to the low pressure of LFRs and SFRs, in the case of a primary coolant leak, core cooling is maintained by a backup structure, such as a guard vessel, without needing a dedicated cooling system. This is different from WCRs, which require coolant injection and depressurization systems. HTGRs using TRISO fuel may not require an emergency cooling system.

Requirements 54–58 refer to the containment structure and containment system. The following gaps were identified:

(af) The containment structure typically used by WCRs is not used by some non-water cooled EIDs. Hence, the requirements related to containment may not consider the loadings and hazards related to the confinement function in these EIDs.
(ag) In some EIDs, such as HTGRs, the TRISO fuel ensures the confinement of radiological substances. Nevertheless, this confinement is not completely leaktight. The allowable leakages of the confinement in operational states

and accident conditions, as well as the monitoring of the leakages, are not considered.

(ah) The approach of early containment venting in the case of a LOCA at HTGRs seems to be in contradiction to the safety approach described in SSR-2/1 (Rev. 1) [3].

(ai) Requirement 57 indicates that access to the containment shall be through airlocks. However, for some SMRs, access to the containment may have different technical solutions that are not covered by this requirement. For example, there could be bolted flanges instead of airlocks.

Requirements 32, 59, 60, 65 and 68 concern the design of I&C systems. The standard does not appear to consider the following aspects:

(aj) Requirement 32: While the text is applicable as written, some EIDs come with novelties relevant to operator performance (see Section 3), and related specific guidance could be considered in explanatory guides (see Section 4.3.7).

(ak) Requirements 65 and 66: While the text is applicable as written, some EIDs come with novelties relevant to control room design and supplementary control room design (e.g. EIDs operated remotely; see also Section 3), and related specific guidance could be considered in explanatory guides (see Section 4.3.7).

(al) Requirement 68: This requirement assumes an emergency power supply and an alternate power supply as essential support systems for safety provisions to control design basis accidents and DECs. EID technology designs could have the following characteristics:
   - They do not require safety grade AC power;
   - They rely more on DC power supply for controlling loss of off-site power and accident conditions, but may need only compartmentalized or even no DC power for some of the functions mentioned in the text;
   - They do not need an alternate power source;
   - They do not foresee a prime mover;
   - They do not need non-permanent equipment except for DC power supply;
   - They have a different electrical power system architecture with reduced levels of redundancy, diversity and capacity.

(am) Requirement 68 also identifies a severe accident with core melt, so its wording does not explicitly include support to provisions necessary to avoid radiological releases to the environment for severe accidents in some EIDs such as HTGRs. The application of this requirement to some EIDs will therefore require at least careful interpretation and possibly abrogation of certain aspects.

Requirements 37, 69–73, 75, 76 and 78, 79 concern the design of auxiliary systems and Requirement 80 concerns the design of fuel handling and storage systems. These are generally applicable to EIDs; however, the following gaps were identified:

(an) Requirement 72 (compressed air systems): This requirement does not include gas systems other than for instrument control.
(ao) Requirement 78 (systems for treatment and control of waste): This requirement may not account for gaseous radioactive waste from EIDs. In particular, two relevant aspects are gaseous radioactive waste coming from the helium purification system in an HTGR and managing polonium in LFRs.
(ap) Requirement 80 (fuel handling and storage systems): This requirement is applicable to EIDs, but it considers only water for cooling spent fuel. The possibility of air cooling or other cooling means, such as sodium, is not included, nor is the storage of the molten salt in MSR concepts. Fuel handling for these concepts, which is made by the transfer and draining of the liquid fuel, is not covered. In addition, EIDs may need additional features to prevent or mitigate potential new failure mechanisms related to non-water coolants and coolant freeze, which may lead to blockage of coolant circulation. The removal of coolant adhered to the fuel for cases of transferal of non-water coolants to a water pool is also not covered.

The following general gaps were also identified:

(aq) In addition, EIDs may introduce new PIEs (see the discussions in Sections 3.2.1.1 and 3.2.1.3), including those that may arise from hazards such as sodium fires in an SFR, or air and water ingress in HTGRs. Sodium burns in air and reacts with water, so water systems need to be avoided in compartments containing, or likely to contain, sodium. Accordingly, novel functions and provisions may be necessary to prevent and mitigate such hazards and PIEs, which could lead to the need for new requirements for systems (see section 6 of SSR-2/1 (Rev. 1) [3]), depending on the importance of these functions and provisions.
(ar) The definitions of ‘controlled state’ and ‘safe state’ in SSR-2/1 (Rev. 1) [3] do not consider EID concepts for which the negative reactivity feedback effect can be sufficient to limit the core temperature for all situations, if the decay heat is removed. Potential examples of these EIDs are HTGRs and

MSRs. In addition, a low power controlled state in an MSR may be enough over a grace period, given that there is sufficient control over the heat sinks. Such a low power controlled state is an additional state, different from the shutdown state.

4.3.2. IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants

4.3.2.1. Areas of applicability

The recommendations and guidance in SSG-30 [32] on how to meet the requirements established in SSR-2/1 (Rev. 1) [3] for the identification of SSCs important to safety and for their classification according to their function and safety significance remain essentially valid for EIDs.

The engineering design rules for items important to safety are identified in SSG-30 and applied on the basis of considerations related to the functions performed by the SSC (for example, single failure criterion, independence, consideration of CCFs, diversity and qualification). However, SSG-30 does not cover these considerations for NPPs, including EIDs. The engineering design rules are provided in the relevant national or international codes and standards, with proven engineering practices and with due account taken of their relevance to nuclear power technology.

4.3.2.2. Areas of non-applicability

No areas of non-applicability were identified.

4.3.2.3. Identified gaps and areas for additional consideration

No gaps or areas for additional consideration were identified.

4.3.3. IAEA Safety Standards Series No. SSG-52, Design of the Reactor Core for Nuclear Power Plants

4.3.3.1. Areas of applicability

SSG-52 [33] can be carefully applied to EIDs, but requires technical interpretation because guidance is drawn expressly from global operating experience for PWRs, BWRs and PHWRs, which are limited to certain fuel mixes and configurations. This is clearly explained in sections 1.5 and 1.6 of SSG-52. For example, the guidance applies directly to fuel rods, fuel assemblies and cladding, whereas some EIDs have an alternative design approach using different means to establish barriers and fuel integrity.

4.3.3.2. Areas of non-applicability

The following areas of non-applicability were identified:

(a) Design extension conditions with core melt may not be applicable to some non-water cooled EIDs, such as HTGRs.
(b) Some of the terminology used in SSG-52 [33], such as ‘fuel rods’, ‘fuel assemblies’ and ‘fuel pellets’, may not be applicable to some EIDs, such as HTGRs using TRISO fuel.
(c) Some parts of SSG-52 (e.g. paras 3.4–3.6, Annexes I and II) present recommendations on design features and characteristics of WCRs, so they may not be applicable to some types of EID. Examples given are also specific to WCRs. Similarly, some paragraphs focus on PHWRs, so they may not be applicable to other reactor designs. Other examples are the following:
    (i) The maximum linear heat generation rate and the void coefficient of reactivity, which are not applicable to the HTGR design, where the power density effectively replaces the maximum linear heat generation rate as a key parameter.
    (ii) The reference to the reactor coolant pressure boundary might not be applicable to some low pressure non-water cooled EIDs, such as SFRs and LFRs; there is a reactor coolant boundary, but it is not pressurized in these EIDs.
    (iii) Activated corrosion products may not be applicable directly to the HTGR design.
    (iv) Critical heat flux is not a suitable parameter for SFRs or LFRs owing to their high boiling point.
(d) The critical heat flux limits and the critical heat flux correlation concept may not be applicable to some non-water cooled EIDs, such as HTGRs, SFRs and LFRs. In addition, heat transfer degradation in fast transients for these EIDs may not be an issue if coolant is retained in the core region.
(e) In normal operation, the neutron flux distribution of some designs of HTGRs and fast neutron EIDs may be relatively uniform, and detailed monitoring of the spatial power distribution may not be necessary. Hence, such monitoring is not directly applicable to these designs.
(f) The concept of the reload fuel cycle and some aspects of core design — such as specification of loading and shuffle patterns of fuel assemblies to provide optimum fuel burnup and desired neutron flux distributions — may not be applicable to some EIDs. For instance, it may not be directly

applicable to EIDs without fuel shuffling. An example is the pebble bed HTGR, as on-line continuous fuel loading and discharging of fuel pebbles is applied. There are also water cooled SMRs that do not use fuel shuffling. For TNPPs refuelled at a central facility, some core design aspects, such as shuffle patterns, might not be applicable. In particular, identifying each fuel assembly and ensuring its proper orientation within the core may not apply to the pebble bed HTGR with coated TRISO particles, since fuel pebbles or spheres cannot be identified and located in the core. However, the position of the fuel blocks of the prismatic HTGR design can be located so this type of identification may be applicable to this design.

(g) For some non-water cooled EIDs, the discussion of mechanical effects in fuel rods and of reactor operation with leaking fuel rods may not be applicable. For an HTGR, a design limit associated with the failure fraction of the fuel particles is used by the designers. Below this design limit, the reactor operation does not distinguish between different values of the failure fraction. In general, fuel rods, fuel assemblies and fuel components of novel design may be assigned to a safety classification according to a graded approach.

(h) Some effects on reactivity, such as the boron reactivity coefficient, may not be applicable to boron free EIDs. Similarly, some systems, features and parameters related to reactivity control — such as control systems using a soluble absorber, in-fuel burnable absorbers, boron addition systems, control rod worth and control bank worth — are not applicable to EIDs that do not use such systems, features or parameters. Guidance about a soluble absorber and a detailed functional analysis of the alignments and operational conditions of the control systems to identify any potential for inadvertent dilution of boron and to ensure the adequacy of preventive and recovery measures is not applicable to boron free EIDs. Some examples related to reactivity control devices are not applicable to some EIDs.

(i) Some types of accident may not be applicable to some EIDs. For example, rod ejection is not relevant to some non-WCRs (e.g. LFRs, SFRs, some HTGRs). Boron dilution is not relevant for EIDs with boron free reactivity control. Some damage mechanisms are specific to WCRs, so they may not be applicable to some EIDs. For instance, flow induced vibration is considered insignificant for the reactor core structures of the pebble bed HTGR. In addition, chemical and volume control systems may not be applicable to SFRs, LFRs or HTGRs.

(j) Some recommendations related to cladding may not be applicable to non-pellet fuels or fuels without cladding, such as those for fuel pellet–cladding interaction. Cladding ballooning is a phenomenon that is specific to fuel rod geometry. For non-pellet fuels, there is no risk of the internal pressure of the fuel rods increasing to cause cladding deformations that would negatively affect the heat transfer between the fuel pellets and the coolant. Owing to the use of alternative fuel types, cladding types or both, issues such as radial gap closure and fuel pellet cracking are not relevant for non-pellet fuel designs. Fuel–cladding mechanical interaction is not applicable to some fuel designs, such as the HTGR fuel design. Pellet–cladding mechanical interaction and stress corrosion cracking may not be applicable to the fuels or cladding of some EIDs, such as the pebble bed HTGR design, some ATF concepts or WCRs with non-pellet fuel. In general, some cladding materials may not be susceptible to hydriding and corrosion. In particular, hydriding does not occur in some non-water cooled EIDs, such as the HTGR, SFR and LFR designs. In addition, some water cooled designs use fuel cladding that is corrosion resistant. For the LFR design, in the event of severe core damage, fuel dispersal to the coolant may be beneficial to prevent fuel compaction and fuel melting.

(k) Hydrogen pick-up correlations are not applicable to cladding materials that are not susceptible to hydrogen pick-up and degradation or adversely affected by corrosion; this may be the case for some non-water cooled EIDs, such as the HTGR, SFR and LFR designs. This may also be the case for some water cooled designs that use fuel cladding that is corrosion resistant. This is because of differences in the hydrogen generated by the chemical reaction between the coolant and the cladding during a LOCA. For reactors using alternative cladding materials, such as ATF cladding, and for non-water cooled EIDs, the amount of hydrogen generated may be a fraction of the amount of hydrogen that would be generated for existing reactors. For existing reactors, it is assumed that all claddings surrounding the fuel pellets in the reactor core react with the coolant, but this assumption may not be applicable for alternative cladding materials and for non-water cooled EIDs.

(l) The guidance provided on moderators may not be applicable to fast reactors, such as SFRs and LFRs.

4.3.3.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) As mentioned in Sections 4.3.3.1 and 4.3.3.2, the terminology using fuel rods and fuel assemblies may not cover all EIDs; for example, it does not cover HTGRs using TRISO fuel. In addition, some designs may use axially movable absorbers and fixed absorbers, or the ex-core, reversed flag type rods used in LFRs. This gap was identified in many paragraphs of SSG-52 [33].

(b) Some parts of SSG-52 [33] (e.g. paras 1.10 and 3.4–3.6, annexes I and II) are focused on WCRs and do not cover other types of EID. For instance, annex I of SSG-52 does not include specific examples for EID cores (e.g. boron free reactor designs), new cladding materials (ATF cladding) or new fuel designs (e.g. cermet material considerations). Annex II does not mention design challenges for new cladding and fuel materials, including HALEU utilization, new fuel and core design features, and related core components and equipment. In general, the rationale for the design, the materials selected and the design characteristics, as well as the performance and functionality (e.g. for categories such as fuel, coolant, moderator and control materials), of EIDs may not be covered.
(c) Section 2 of SSG-52 [33], which focuses on general safety considerations in the design of the reactor core, does not mention the core and fuel design features of EIDs that aim to improve safety performance.
(d) Regarding the safety classification aspects of the reactor core, the guidance is written for core designs based on fuel assemblies and control rods. Appropriate interpretation (at least) would be needed to apply the recommendations to alternative core and fuel designs.
(e) Significant fuel degradation is claimed to be precluded in some non-water cooled EIDs, such as the HTGR. The fuel design, performance and safety approach of these non-water cooled EIDs are different to those of current WCRs. For example, the barrier concept used in the HTGR multilayer coated particle design is claimed to provide a confinement barrier to prevent fission product release in normal operation and accident conditions without significant core degradation.

Section 3 of SSG-52 [33] has the following gaps:

(f) Regarding the fuel type, the specificities of metallic fuel for SFRs (such as degradation mechanisms) are not included. For fast neutron reactors, such as SFRs and LFRs, reactivity feedback due to core and fuel deformation may be credited if it results in reactivity decrease, but it is not included.
(g) SSG-52 does not cover some reactivity aspects of non-water cooled EIDs. For example, the inherent reactivity feedback (e.g. Doppler feedback, fuel axial expansion, core radial expansion), passive reactivity feedback and passive reactivity reduction mechanisms are key SFR and LFR design features to prevent core damage during accidents, especially during DECs before core melt.
(h) While the principles relating to thermohydraulic design features (e.g. paras 3.30–3.33 of SSG-52) are generally applicable to WCRs, and therefore to water cooled EIDs, developments such as ATF and non-water cooled EIDs

can include new fuel and cladding materials. These materials, as well as research, development and testing, are likely to affect the analysis approaches, but are not covered. In addition, the specific thermohydraulic design limits may not address those for non-WCRs, where peak fuel temperature and enthalpy are key indicators. For the SFR and LFR designs, typical metrics are the maximum temperature of the cladding tube or of the coolant. Limit values are set below the coolant boiling temperature with sufficient margin. In general, recommendations in several subsections of section 3 of SSG-52 also have gaps of a similar nature to those for thermohydraulic design.

(i) A similar gap to that mentioned in (h) also applies to the thermomechanical design of fuel rods and fuel assemblies (e.g. paras 3.34–3.39 of SSG-52); for instance, in relation to differences in cladding material for SFRs. Deformation with power may be credited as a safety feature for metallic fuel, and the effects of axial movement of the fuel material may be similarly credited for certain accident conditions. The movement and handling of pebbles are considerations for pebble bed HTGRs.

(j) Stress corrosion cracking for materials other than fuel cladding, including any core components or structural materials supporting or affecting the core, is not addressed.

(k) The guidance on the mechanical design of core structures and components is very focused on WCRs, but in most cases, high level objectives would be applicable to EIDs. Guidance on specific technologies is not provided. Examples of gaps are the following:
   (i) For lifetime cores or cores with very long periods of time between refuelling, the limited capacity to inspect and replace components, including reactivity control mechanisms, may pose a challenge.
   (ii) For some reactor technologies, chemical control may become critical to ensuring continued fitness of mechanical components, fuel or both; an example is the oxygen control in LFRs.
   (iii) Chemical control could also be affected by contamination (e.g. oxygen ingress in SFRs), water contamination or hydrogen contamination for SFRs (from the secondary side), and oil contamination (pump bearings). Provisions for cleaning may need to be made in the design.
   (iv) The impact of freezing of the coolant on SSCs needs to be considered.

(l) The approach to maintaining fuel integrity and assuring acceptable radiological conditions in normal and accident conditions for non-water cooled EIDs may not be covered. In general, the principles, limits and examples in some parts of SSG-52 [33] may not address EIDs. Specifically, for non-water cooled EIDs, as well as novel fuel concepts for WCRs, including some ATF concepts, the considerations of damaged fuel and its handling in the design are not covered.

(m) Aspects of core cooling of non-water cooled EIDs may not be covered. Two examples of these gaps are: (i) fuel melting or the areal melt fraction is used for the fuel design limits of the SFR and LFR designs, and (ii) the HTGR design identifies the fuel temperature margin as the criterion for fuel failure evaluation.

(n) The core (support) structures of non-WCRs may not be addressed. For example, SFRs and LFRs have core support structures that differ from those of the WCR designs.

(o) In most cases, the high level objectives of the reactor core control system would be applicable to EIDs. However, no additional guidance is provided on the rules and acceptability of reliance on inherent reactivity feedbacks; for example, power control by load control on the secondary side and the resulting temperature reactivity feedback on the core. Some fast neutron EID developers consider, for example, that inherent reactivity feedback and passive reactivity reduction mechanisms are key to reactor shutdown and preventing core damage. In addition, some advanced reactor designs rely primarily on temperature feedback for effective reactivity control and reactor power level manoeuvres, with some use of mechanical reactivity devices. A strong negative temperature reactivity coefficient is identified as a second means of shutdown for some EIDs, but it is not covered. The acceptable implementation and related considerations are not discussed; similar guidance would be useful for newer BWR designs. Guidance on, and consideration of, treatment of the delayed neutron fraction in the design, operation and safety analysis for some EIDs is also not included.

(p) For most cases, the guidance on the reactor shutdown system has high level implied objectives that would be applicable to EIDs. However, it does not cover rules and acceptability of designs that rely on the use of inherent reactivity control mechanisms as a reliable means of shutdown (e.g. for HTGRs). In particular, it does not include a grace time to establish a guaranteed shutdown state when there is potential for core re-criticality after a period without direct intervention. For example, current guidance relies heavily on reactivity control and shutdown systems that are rod based. In addition, no guidance is provided on defining appropriate parameters that will govern the acceptability of the shutdown system capability or on demonstrating that a suitable state of subcriticality has been established. Moreover, the consequences for the core and reactor design (e.g. reactivity feedback effects) are not considered with respect to unprotected transients (i.e. design extension conditions related to PIEs combined with the failure of shutdown systems).

(q) Some EIDs may incorporate higher fuel enrichment levels than current reactors. This might be an additional non-proliferation consideration to be considered when discussing safety, security and safeguards by design (see Section 5).

(r) The guidance on core management is very focused on WCRs, but in most cases, high level implied objectives would be applicable to EIDs. However, guidance for some EIDs is not provided. Specifically, guidelines for designs with very long cycle length or lifetime cores that make extensive use of long term reactivity enabling and control mechanisms, such as absorber rods and burnable poisons, are not fully included. In addition, recommendations (e.g. para. 3.141 of SSG-52) on core management programmes for designs without partial refuelling and fuel shuffling may not be covered.

(s) Parameters associated with depletion of fuel and the burnable absorber, as well as other reactor physics parameters, are provided as inputs to safety analyses, plant monitoring and protection systems and operator guidance. However, non-water cooled EIDs will likely employ advanced plant monitoring and protective system instrumentation and measurement protocols, and they may have differing approaches to thermal to fission power calibration. These aspects are not included.

(t) The guidance on the design of a MOX core focuses on WCRs. Aspects that are not covered are: (i) among non-WCRs, MOX fuel with high plutonium levels, used in LFRs and SFRs; (ii) high breeding ratio reactors, including those with breeding ratios >1 (i.e. using breeder blankets); and (iii) some fast neutron EIDs using fuel with minor actinides.

(u) Zr based alloy cladding and its impacts on used fuel handling and storage are referenced, but other types of cladding for use in non-water cooled EIDs and some current or proposed WCRs, such as ATF, are not covered.

(v) HTGR fuel is not fully covered.

(w) Post-irradiation activities, including fuel transport, storage and reprocessing of the fuel, are not fully covered. Some related gaps are the following:
   (i) Non-water cooled EIDs, such as HTGRs, are not covered.
   (ii) Issues related to a closed nuclear fuel cycle in fast neutron EIDs, including the use of Pu from irradiated MOX fuel in WCRs, are not covered.
   (iii) Although a recommendation to address the back end of the nuclear fuel cycle is provided, no details are included.

(x) Section 4 of SSG-52 includes considerations for prototype and lead use assembly testing. The testing plans and protocols for EIDs may differ from programmes developed and instituted for the WCR fleet. For example, in-reactor test programmes for fuel assemblies designed for fuel dwells of 10–20 years may be impractical. In general, the types of test to be performed for the different types of fuel and new materials used in EIDs may not be addressed.

(y) There are guides that address fuel handling, fuel management and the safety of uranium fuel fabrication facilities for EIDs, but there appears to be no guidance that addresses considerations in development, assessment and quality assurance for fuel manufacturing for novel fuels. Given the importance of fuel quality and integrity in supporting DiD provisions for EIDs, guidance for assessment and control of manufacturing methodologies and activities is important to support safety assessments for these proposed reactor facilities. These aspects, however, do not seem to be explicitly covered in current IAEA safety standards.

4.3.4. IAEA Safety Standards Series No. SSG-56, Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants

4.3.4.1. Areas of applicability

SSG-56 [34] can be carefully applied to EIDs, but requires technical interpretation because guidance is drawn expressly from global operating experience for PWRs, BWRs and PHWRs. Paragraph 3.1 states that section 3 of SSG-56 is applicable to all WCRs.

In general, recommendations in paras 3.60–3.75 that relate to the prevention of abnormal conditions from escalating to accident conditions with an adequate reliability are applicable to the identified areas of novelty.

The elements for designing a safety system are not reactor technology or design dependent. For example, the reliability of a system is driven by the safety significance of the function(s) that it accomplishes, and therefore does not have such a dependence. Hence, these elements are applicable and are addressed in section 3 of SSG-56.

Paragraphs 4.2–4.19 focus on the adequacy of the UHS and its availability to absorb heat, including decay heat, and release heat to the environment. These paragraphs can be considered generally applicable to land based EIDs.

Paragraphs 5.2–5.17 mainly focus on structural design, design basis loads and load combinations and can also be considered applicable to EIDs.

4.3.4.2. Areas of non-applicability

The following recommendations are technology specific and therefore not fully applicable to some EIDs:

(a) Some items are non-applicable to non-water cooled EIDs, for instance paras 3.22 and 3.28 include examples that are not applicable to these EIDs.

(b) Paragraphs 3.107–3.114 are specific to the in-service inspection of pressure retaining components and hence are not applicable to non-pressurized reactors.
(c) Regarding para. 3.137 on multiple units at a site, the applicability has been discussed at the level of SSR-2/1 (Rev. 1) [3], Requirement 33 (see Section 4.3.1).
(d) Paragraphs 4.20–4.45 are relevant to DHR in operational states and in accident conditions. Most of them[24] can be considered as applicable to other reactor technologies. However, for EIDs with a low core decay heat, the heat transport to the UHS could be simplified by relying on mechanisms of natural conduction, convection and radiation, which would make the recommendations provided in section 3 of SSG-56 [34] inapplicable because the decay heat is not removed by systems. Decay heat removal by natural convection may also be relevant for liquid metal cooled EIDs (SFRs and LFRs) with power greater than that of SMRs.
(e) Paragraphs 5.1 and 5.18–5.51 address pressure control, overpressure protection and isolation of the reactor coolant pressure boundary. As currently phrased, they are typically oriented to pressurized WCRs. Such guidance would not be very useful for reviewing the design of items controlling the pressure in non-pressurized reactors; hence, they may not be applicable to these reactors.
(f) Paragraphs 5.52–5.76 are not applicable to non-pressurized reactors.
(g) Paragraph 5.54 addresses PIEs and provides typical examples, such as loss of off-site power. However, a floating TNPP may not have an off-site power supply, except when it docks to provide electricity to the land. Hence, loss of off-site power does not appear applicable to this type of NPP when it is not connected to an off-site power supply.
(h) Paragraphs 5.79–5.127 address specific design aspects of the RCS components of WCRs and hence are not applicable to non-pressurized reactors.
(i) Section 6 of SSG-56 is specific to PWR technology, for example, discussing how the boric acid concentration of the primary coolant is adjusted. However, some EIDs may have a boron free core with only an emergency boron injection subsystem. Hence, much of the guidance in this section may not be applicable to these EIDs.
(j) Sections 6–8 of SSG-56 offer practical guidance and recommendations that explain the ways in which systems can be made compliant in practice with the design requirements for the current water based technologies (PWR,

[24] Recommendations dealing with specific design aspects are oriented to large pressurized WCRs.

BWR, PHWR). However, these sections do not provide recommendations that are specific to EIDs, such as which PIEs might need to be considered and how their consequences might be mitigated.

4.3.4.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) Non-WCRs can have accident conditions that influence reactor coolant systems not covered in para. 3.28 of SSG-56 [34] (e.g. breach of normal chemical regime, freezing, ingress of secondary circuit water, steam into primary circuit coolant).
(b) For some non-WCRs, the examples of parameters in para. 3.133 of SSG-56 may not be complete. For example, chemical parameters (such as oxygen concentration) are not covered.
(c) Systems that are designed to mitigate accident conditions may be specific to each reactor technology. Hence, the specifics of these systems may not be addressed in section 3 of SSG-56.
(d) Paragraphs 4.2-4.19 of SSG-56 on the UHS aim at securing its availability and minimum capacity. For modular reactors, guidance has to be applied on the module or unit level, as appropriate, for adequate results, but this is not mentioned.
(e) Paragraph 4.19 of SSG-56 is related to non-radiological heat sources due to possible chemical reactions in accident conditions. For WCRs, the main possible reaction is between water and cladding metal. For non-water cooled EIDs, other chemical reactions could occur that are not considered.
(f) The failure modes, mechanisms, loading conditions and hazard loads that challenge the integrity of the components of the primary circuit of EIDs may not be adequately covered in section 5 of SSG-56; for example the following:
    (i) The failure modes and mechanisms indicated in paras 5.3–5.10;
    (ii) The loads to be considered in the analysis presented in paras 5.11–5.16;
    (iii) The adequacy of the recommendations to control and limit the primary pressure described in paras 5.20–5.43;
    (iv) Non-pressurized reactors (paras 5.52–5.76);
    (v) The internal hazards (paras 5.55 and 5.56) that could challenge the integrity of RCS equipment.

The main areas of novelty and specific characteristics of EIDs are the operating pressure and temperature, the sensitivity of the materials used to failure modes (e.g. excessive plastic deformation, buckling, progressive deformation and ratcheting, fast and brittle fracture, thermal fatigue, corrosion), the neutron fluence and the nature of the primary coolant (e.g. chemistry, high heat capacity of the coolant, high heat transfer capacity of the coolant). Some of the PIEs are also specific to a reactor technology or to a design and therefore may influence the recommendations for the design of systems. However, these areas of novelty may not be covered by SSG-56 [29]. Hence, the guidance may not fully consider the failure modes, mechanisms, loading conditions and hazard loads of EIDs that challenge the integrity of the components of the primary circuit.

4.3.5. IAEA Safety Standards Series No. SSG-53, Design of the Reactor Containment and Associated Systems for Nuclear Power Plants

4.3.5.1. Areas of applicability

Guidance in SSG-53 [35] is generally derived from the operating experience and safety characteristics of WCR configurations. This includes simultaneously addressing high pressures and other energetic internal events associated with the fuel types, types of initiating events and use of water as a coolant, in conjunction with internal and external events.

Section 1 of SSG-53 provides introductory information and is mainly applicable to reactor concepts of the WCR type. Section 2 is fully applicable to reactor concepts of the WCR type and mostly to non-WCR types. Sections 3 and 4 are mostly applicable to reactor concepts of the WCR type but require significant interpretation to apply to EIDs (exceptions are discussed below).

Paragraph 4.116 is restricted to indirect cycle heat transport arrangements with steam generators, such as PWRs. Direct cycle heat transport concepts, such as BWRs, have no steam generators and, consequently, SGTRs need not be considered. Accordingly, this paragraph may be applicable to EIDs with steam generators[25]. It is also applicable to HTGRs, except for those concepts with a direct cycle utilizing a helium turbine.

Section 5, which is related to in-service tests and inspections, is applicable to EIDs, including those that are fully or partially embedded underground.

[25] An SGTR is typically a leak from the primary to the secondary circuit due to the difference of pressures between the primary and secondary circuits in a steam generator. Even if the reactor vessel is submerged in a pool of water, as designed in some EIDs, radionuclides transferred from the primary side can be transported by the steam to the turbine and, thus, to a location outside the affected unit or module. Hence, this may be a bypass scenario in which the release may not be stopped by the pool water in designs with the reactor pressure vessel submerged in such a pool.

4.3.5.2. *Areas of non-applicability*

The following areas of non-applicability were identified.

General:

(a) SSG-53 [35] provides many recommendations that assume the most severe accident in a large WCR to be a core melt scenario. This is rooted in the specific wording in SSR-2/1 (Rev. 1) [3]. For several EIDs, a severe accident may not be related to the core melt terminology. For TRISO fuel, for example, it may be possible to show that core melt is practically eliminated. If so, other, more adequate severe accident conditions may be applicable. In addition, SSG-53 implicitly assumes that the containment system will resemble that of current light water reactors, whereas a different concept for the final confinement barrier or containment might be realized in several EIDs, such as HTGRs. In such cases, many assumptions behind the recommendations of SSG-53 may not apply. However, the containment represents the last barrier to prevent releases to the environment in a current WCR, and reliable barriers must be provided for an EID, even for the most challenging accident conditions postulated for a reactor design. In particular, a structure may have to be provided to protect from external hazards. Consequently, at least a careful interpretation of the requirement objectives and a graded approach regarding several EIDs seems advisable.
(b) SSG-53 provides recommendations fully adapted for WCRs in which the containment is a strong single or double structure that provides three safety functions: (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. In non-WCR designs, the three safety functions remain but they are supplied by different design features. For example, in HTGRs, the confinement of radioactive substances is mainly provided by the TRISO fuel, and the protection against external hazards is achieved by a concrete structure, such as the reactor building, which does not have the same requirements as the containment structure of WCRs. Typically, the HTGR reactor building may not need to be designed for high pressures and temperatures and does not have to meet the same requirements for leaktightness as those applied to WCR containments. Radiation shielding is also generally performed with different features. For example, radioprotection of workers is achieved by features implemented in the containment near the radioactive sources. Moreover, for non-WCR designs, the internal loadings to consider for the design of these features are

of a different nature from those typically considered for WCRs (e.g. hazards induced by sodium in SFRs).

The following are specific areas of SSG-53 [35] that may not be fully applicable to some EIDs. However, some of these items are also related to the gaps mentioned in Section 4.3.5.3.

Postulated initiating events:

(c) Paragraph 3.8 lists typical PIEs to be considered for the design of the containment. The PIE list is generally applicable although fuel handling accidents may not be applicable to factory fuelled reactor designs (i.e. without fuel handling equipment).

Accident conditions:

(d) Paragraph 3.31 recommends that the design of the containment and its systems is such that venting is not necessary in design basis accidents. However, the reactor building will be vented in case of LOCAs in an HTGR to release He, since it is a non-condensable gas. Hence, this recommendation may not be applicable to HTGR designs.
(e) The pressure buildup addressed in para. 3.37 might not be applicable to some non-water cooled EIDs, such as SFRs and LFRs.
(f) Paragraph 3.41 recommends the postulation of core melt accidents as DECs in line with SSR-2/1 (Rev. 1) [3]. This would not be applicable to EID concepts where core melting is not a relevant accident scenario.
(g) Paragraph 3.42 is related to the design of safety features in WCRs in case of DECs with core melting, and it provides aspects that may affect the accident progression and that may influence the containment response and the source term. It is not applicable to non-WCR designs that preclude a core melting scenario, such as HTGRs. In addition, some of the aspects described in that paragraph might be inapplicable to these EIDs for other reasons, such as the different characteristics (compared with WCRs) of the conditions within the containment in case of a major accident.
(h) Paragraph 3.44 concerning pressure buildup and venting to ensure containment integrity may not be applicable to some non-WCR designs, such as SFRs and LFRs. In particular, venting is considered a mitigation measure in case of a LOCA in an HTGR.

Reliability:

(i) Paragraph 3.54, recommending an emergency power source for mass and energy release and management, may not be applicable to EIDs with completely passive safety systems.
(j) Paragraph 3.59, recommending a diverse or alternate power source for DECs without significant fuel degradation, may not be applicable to EIDs with completely passive safety features. Notably, this is a consequence of the specific wording in para. 6.44B of SSR-2/1 (Rev. 1) [3]; see para. 3.61 of SSG-53.

Practical elimination of conditions that could lead to an early radioactive release or a large radioactive release:

(k) Paragraph 3.67 (b) of SSG-53 [35] addresses core melt accidents with a containment bypass, and it may not be fully applicable to EIDs that do not have core melt accidents.
(l) Paragraph 3.68 is specific to severe accident phenomena in WCRs, so it may not be applicable to non-water cooled EIDs.

Design of the containment and its associated systems:

(m) Paragraph 4.4, items (a) and (h), might not be applicable to some non-water cooled EIDs, such as SFRs and LFRs. For example, non-WCRs do not have sump operation for emergency core cooling; also, large mass and energy release is a typical phenomenon for WCRs but is not limited to WCRs.
(n) Paragraph 4.13 may not be fully applicable to multi-module concepts (such as SMRs) having several reactors that share, for example, a common reactor pool. In general, it may not be applicable to EIDs having shared systems.

Structural design of the containment and of structures within the containment:

(o) Paragraphs 4.40, 4.52 and 4.57 may not be applicable to floating TNPPs. SSG-53 assumes that an SL-2 earthquake is site specific because the Safety Guide is written for stationary, land based WCRs, but this assumption may not be applicable to floating TNPPs, because they are expected to be more impacted by acceleration forces due to waves.

Mass and energy release and management:

(p) Paragraph 4.60 deals with the ventilation system, so it is applicable only to designs having a 'gaseous' containment atmosphere. Accordingly, it may not be applicable to designs with containment vessels immersed in water pools, such as integral PWR designs.
(q) Paragraph 4.65 deals with water stored within the containment. Specifically, this paragraph proposes to evaporate and condense water to remove heat. This necessitates the presence of sufficient water in the reservoirs. However, there are not necessarily large water reservoirs within the containment in non-water cooled EIDs, so this paragraph may not be applicable. The recommendation in para. 4.66 concerning the containment spray system may not be applicable to non-water cooled EIDs, because this system would not help in the case of non-condensable gases, such as helium, and water may not be used with coolants that chemically react with it, such as sodium.
(r) The pressure suppression pool systems recommended in paras 4.71–4.79 may not be applicable to non-WCR designs, because such systems are mainly designed to condense steam in WCR concepts.
(s) Paragraph 4.82 may not be applicable to EIDs with passive safety features relying on natural convection.
(t) Paragraphs 4.84 and 4.85 may not be applicable to non-WCR designs, because sump operation for emergency core cooling may not be possible or foreseen in these designs.

Control and limitation of radioactive releases:

(u) Paragraph 4.119 is very specific to WCR concepts. In particular, items (b) and (c) may be not applicable to non-WCR designs.
(v) Paragraph 4.122 may not be applicable to non-water cooled EIDs, because the containment spray system would not help in the case of non-condensable gases such as helium.
(w) Paragraph 4.124 deals with suppression pools, so it may not be applicable to non-water cooled EIDs, such as SFR, LFR and HTGR.

Management of combustible gases:

(x) Regarding para. 4.133 of SSG-53 [35], the following apply:
    (i) Items (a), (b), (e) and (f) may not be applicable to non-WCR designs, because: regarding item (a), no water in the core is expected for these EIDs (except perhaps in accident conditions); regarding item (b), these EIDs will have neither a sump nor a suppression pool; regarding

item (e), degassing of hydrogen is important only for reactor concepts using water as primary coolant; and regarding item (f), hydrogen tanks are important for WCRs to adjust, for example, the pH of the coolant, but this is not necessary for non-water coolants.
    - (ii) Item (g) is not applicable to some TNPPs that lack concrete structures (some floating TNPPs have concrete structures), because molten core–concrete interaction will not occur. However, interactions between corium and the support structures (e.g. steel) may happen.
- (y) Paragraph 4.141 is specific to WCRs and may not be directly applicable to non-WCR designs.

Mechanical features of the containment:

- (z) Paragraphs 4.175–4.178 may not be applicable to small containments where no access is necessary during operation, because such containments may have permanently closed openings (e.g. bolted flanges).

Materials:

- (aa) Paragraph 4.191 is very specific to WCRs and might not be applicable to non-water cooled EIDs.
- (ab) Paragraphs 4.196 (a), 4.197 and 4.200 may not be applicable to EIDs without sumps or without sump recirculation.

Instrumentation:

- (ac) Paragraph 4.211 recommends seismic instrumentation. This paragraph may not be applicable to floating TNPPs after deployment, because they are not directly affected by earthquakes, but by acceleration forces, such as waves. It may be relevant if they are deployed close to land.
- (ad) Paragraph 4.221 may not be applicable to non-WCR designs. Parameters other than humidity may be relevant for indicating a leak from the primary circuit for these concepts.
- (ae) Regarding paragraph 4.237, the following apply:
    - (i) Item (a) may not be applicable to some non-WCR designs, such as SFRs and LFRs, because these reactors are operated with low primary coolant pressures.
    - (ii) Item (d) may not be applicable to EIDs without containment spray systems.
- (af) As to the multiple recommendations identified above relating to seismic hazards, which are not applicable to floating TNPPs, they might become

relevant if these designs are located close to land. In this case, interpretation and a graded approach will have to be applied.

4.3.5.3. Identified gaps and areas for additional consideration

The following gaps were identified in SSG-53 [35]:

(a) In contrast to the transportation of fresh fuel assemblies, a factory fuelled TNPP may have a reactor in operation (cold shutdown). TNPP concepts investigated so far have dedicated steel shell or compact containment structures, but other concepts may be possible. In any case, parts of the overall containment system, such as structures protecting against external hazard impacts, might be absent during transport. These aspects are not discussed in SSG-53. Similarly, there is no guidance on temporary containment provisions during transport, if applicable. A question to be resolved is whether this aspect is part of the reactor design or a requirement for transportation.

(b) In addition, a TNPP or one of its modules (in the case of a modular design) may need to be integrated into a containment system at separate, possibly off-site, buildings for servicing and refuelling. Such containment systems might be temporary. The impact that these aspects could have on containment design is not addressed.

(c) For EIDs for which core melting accidents are precluded by design (e.g. HTGRs with TRISO fuel), a surrogate definition of the term 'severe accident' might be required to develop the associated recommendations.

(d) EIDs using a functional containment approach, such as HTGRs with TRISO fuel, are not covered. Providing recommendations for EIDs using a functional containment approach would have the goal of achieving a safety level at least similar to that obtained with the containment system of WCRs, in terms of efficiency and reliability. The containment system of WCRs contributes to controlling the release of radioactive material from the plant (e.g. by keeping the pressure in the containment below atmospheric pressure in normal operation). This function needs to be maintained for all EIDs, including those using a functional requirement approach. The demonstration of practical elimination of a containment bypass is also not covered for EIDs using a functional containment approach.

(e) Paragraph 1.5 limits the scope of SSG-53 [35] to land based, stationary WCRs, so it does not include many EIDs. Moreover, fully or partially embedded EIDs are not completely covered.

(f) Paragraph 3.8 lists typical PIEs to be considered for the design of the containment. The list of PIEs may not cover PIEs for EIDs.

(g) Paragraph 3.24 lists some loads to consider for the design of the containment and its associated systems. However, the most representative new loads likely to occur during accidents in non-WCR designs are not included. For example, for SFRs, there is no consideration of the impact of sodium release on concrete or steel containment, such as the effects of high temperatures, sodium fires, chemical reaction of sodium with concrete and products released from sodium–air or sodium–water reactions, such as hydrogen and sodium hydroxide. Durability over design life is not considered either.
(h) Paragraph 3.36 addresses three types of failure to be considered as DECs in WCRs. However, it does not consider accident conditions affecting several modules at the same time as a subset of multiple failure events in the case of modular EIDs, or other failure events or multiple failures for EIDs, and their impact on the integrity of the containment.
(i) Paragraph 3.43 is applicable in general to all types of reactors, although it focuses on phenomena of severe accidents in WCRs. Hence, it does not discuss or appropriately reflect the relevant phenomena of non-water cooled EIDs, such as SFRs (e.g. risk for a sodium fire) or LFRs.
(j) Paragraphs 3.88–3.90 address PSA, but they do not discuss multi-unit considerations, nor multi-module aspects (such as those of an SMR), since each module may have one reactor.
(k) Paragraph 4.3 addresses a current WCR containment but does not include other reactor designs, such as pool-type reactors, TNPPs or underground reactor installations.
(l) Paragraph 4.22(a) addresses mass and energy release, for example, from metal–water reactions. These phenomena are specific to severe accidents in WCRs. However, this clause does not discuss potential chemical reactions in non-water cooled EIDs.
(m) Paragraph 4.46 addresses failure modes of the containment. However, this paragraph does not discuss new failure modes that may need to be considered for EIDs.
(n) Paragraph 4.51 provides recommendations to remove the heat of the molten core in the case of in-vessel retention during an accident. The recommendations are developed for WCRs (e.g. need to flood the reactor cavity) but non-water cooled EIDs that consider also in-vessel retention, such as some SFRs, are not discussed.
(o) Paragraph 4.133 does not consider some sources of combustible gases for non-WCR designs (e.g. formation of carbon monoxide in the case of air or water ingress accidents in HTGRs).
(p) Regarding paras 4.181–4.188, which address materials, some designs, such as transportable SMRs, may use steel instead of concrete for the containment. However, using steel or other materials is not considered.

(q) Regarding paras 4.196(a), 4.197 and 4.200, debris may challenge the efficiency of passive heat removal, but this is not specifically mentioned regarding EIDs.
(r) Regarding para. 4.211, floating TNPPs may need sensors to measure acceleration forces, for example, due to waves. However, this is not considered.
(s) Paragraph 4.221 may not be applicable to non-WCR designs. Factors other than humidity may be highly significant for indicating a leak from the primary circuit for these concepts, but they are not covered.
(t) Paragraph 4.225 recommends activity measurements in water drain storage or sumps. Suitable locations for such measurements are different for reactor concepts without such storage or sumps, but they are not covered.
(u) Paragraph 5.3 is applicable to traditional concrete containment designs. However, it does not provide guidance for steel containment vessels or for containments for reactors located on barges or ships.

4.3.6. IAEA Safety Standards Series No. SSG-34, Design of Electrical Power Systems for Nuclear Power Plants

4.3.6.1. Areas of applicability

SSG-34 [36] is considered generally applicable to EIDs, considering that several technical areas may require additional guidance as discussed below. The following two points are noteworthy:

(a) Paragraph 6.15 recommends that a single transmission line for each off-site power supply may be acceptable if the safety analysis report shows that this arrangement achieves the technical safety objectives as defined in SSR-2/1 (Rev. 1) [2]. Specifically, a single off-site power supply might be acceptable for EIDs employing passive engineered safety features, according to a graded approach.
(b) The guidance on DC power systems (paras 7.83–7.127) is applicable to the design of DC power systems of EIDs.

4.3.6.2. Areas of non-applicability

The following areas of non-applicability were identified:

(a) The electrical power supply systems of EIDs with passive safety features are generally similar to those of conventional WCRs albeit without the requirement for the provision of safety classified AC electrical power supplies needed at the onset of an anticipated operational occurrence or accident condition to maintain the plant in a controlled state.

(b) The architecture of electrical power systems and sizing of electrical components (e.g. motor operated valves, electrical motors, circuit breakers, DC power supply systems such as batteries and battery chargers) may not require the same level of redundancy, diversity and capacity to supply necessary loads to bring the plant to a safe state and maintain long term heat removal. There is more emphasis on the design and reliability of DC power systems, including the associated uninterruptible power supplies (e.g. single failure criterion, independence, diversity) as described in SSG-34.

(c) SSG-34 indicates that controls for on-site power systems have to include automatic selection of an alternative off-site power supply when the normal off-site power supply is not available. If a floating TNPP is connected to an off-site grid, a power line from this grid may be considered an alternative off-site power supply, and this SSG-34 guidance would be applicable. However, if such power line is not available to the floating TNPP, this guidance may not be applicable.

4.3.6.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) Design basis for electrical power systems (paras 4.1–4.10 of SSG-34 [36]). This guidance is applicable to the electrical design of an EID. However, there appears to be some gaps, such as the following:
  (i) When references are made to the main generator. In particular, an SMR with multiple modules is comprised of several reactors, and each may have its own generator; the outputs from these generators may be interconnected to a single external grid connection. The variations in the external grid, as defined in section 4 of SSG-34, will apply to each generator. Consequently, this guidance may not cover the measures needed to prevent a disturbance on a single generator from adversely impacting the other modules.
  (ii) For SMRs with multiple modules. In this case, the guidance does not mention that, regarding the design of the AC power supply system: (1) sufficient independence of AC supplies may be necessary so that no single design basis event can result in the total loss of AC supplies to a single module; and that (2) essential facilities on each module may be provided with sufficient redundancy to ensure that no essential system will be lost because of a single electrical fault.

(iii) The guidance does not mention the following:
  - A reliable backup AC power source may be needed to supply a multiple module EID in the event of a loss of off-site power.
  - Redundancy may be required in the form of backup generators and their associated distribution systems so that supplies to individual modules will not be lost owing to system faults, maintenance operations or both.

(iv) The guidance does not mention the following:
  - If construction and commissioning of additional modules takes place on a site with modules operating at power, all construction activities will need to be carried out so that there is no impact on the independence of the power supplies to the operating modules.
  - Electrical connections of new modules to the electrical power supply system may not be made when any individual module is operating at power.

(b) Anticipated electrical events (paras 5.1–5.7). The existing guidance is generally applicable to EIDs. However, there may be some gaps, such as the following:
   (i) Paragraph 5.3 refers to the main generator and may be applicable to EIDs except that, in the case of EIDs with multiple modules, the evaluation is more complex and the actions required in response to a perturbation that has the potential to cause a generator trip are not included.
   (ii) Paragraph 5.4 is applicable, but reference to figure 4 of SSG-34 may not be representative of the electrical systems of EIDs.
   (iii) Paragraph 5.5 does not consider the approach to be adopted for multiple generators.

(c) Station blackout (paras 5.8–5.10). These recommendations may not cover the electrical design of some EIDs with passive systems that do not need near-continuous AC power. For example, a station blackout caused by a loss of the preferred supply concurrent with a turbine trip may not place immediate demands on a backup AC power supply, as the passive design may initially be dependent only on the operation of DC systems.

(d) The design guidelines for electrical safety power systems (paras 7.1–7.127 of SSG-34 [36]) do not cover the following:
   (i) The following considerations are not covered in paras 7.2–7.6: EIDs provided with passive safety features may be provided with DC power systems to supply necessary power to I&C systems in the event of loss of the preferred power supply. The DC power systems may have sufficient capacity to provide power supplies, so that the plant is

maintained in a controlled state until off-site power supplies can be restored or alternative on-site AC supplies can be provided.

(ii) The following considerations are not addressed in paras 7.7–7.19, since the recommendation in para. 7.9(b) will not apply: an NPP will lose external AC power in the case of a loss of off-site power event, and plants with passive safety features may manually activate standby AC power sources to ensure continuity of DC power supply (e.g. beyond the battery discharge time). The standby AC power sources may be independent for each safety division and may be capable of being connected to supply the DC power systems beyond the defined design operating time of the DC power systems. These sources may have the capability to power plant safety systems to maintain the plant in a controlled state and to establish a safe state in accordance with the plant safety case.

(iii) The following considerations are not included in para. 7.15: each division may have independent detection and protection features to disconnect the safety buses from the preferred power supply to shed loads from the safety buses and, where required for active safety systems, to start the standby AC power sources in the event of degradation or loss of voltage or degradation in frequency.

(iv) The following considerations related to maintenance on EIDs with passive safety features are not covered in paras 7.30–7.32: application or non-compliance with the single failure criterion during testing and maintenance activities of the electrical power supply system on plants with passive safety features.

(v) Paragraph 7.36 refers only to standby AC power sources. Accordingly, the following considerations are not addressed: (1) standby power sources may consist of separate AC and DC power supply sources; (2) AC sources may consist of an electrical power generating module complete with all auxiliaries and a dedicated separate and independent stored energy supply for both starting and running the prime mover; and (3) DC sources may consist of an independent DC power system with the capacity to supply all connected loads for the designated operating time and with facilities (e.g. charging batteries) to supply the loads and support operation of the systems needed for long term post-accident management.

(vi) The following considerations related to standby power sources on EIDs with shared facilities are not included in paras 7.35–7.63, as para. 7.37 may not apply: when EIDs with passive safety features are provided with shared standby power supplies covering multiple modules or divisions, detailed safety assessments are important to demonstrate: (1) the adequacy of the power supplies to support plant safety; (2) that an event on one module does not adversely impact the provision of power supplies to other modules; (3) that the single failure criterion is met on all modules during maintenance activities; and (4) that the capacity of the standby power sources is adequate to maintain power supplies to safety systems during all anticipated events for all plant states, including the performance of maintenance activities.

(vii) Paragraph 7.45 does not apply to EIDs with passive safety features, so the following considerations are not covered in paras 7.35–7.63: for EIDs with passive safety features, the features to initiate and replenish passive safety systems and I&C systems may be automatically supplied by DC power systems upon the loss of the preferred power supply. Standby AC power components may be manually activated to supply the DC power systems beyond the defined DC power operating time.

(viii) Paragraph 7.46 does not apply to EIDs with passive safety features, so the following considerations are not included in paras 7.35–7.63: for EIDs with passive safety features, an automatic initiation of the DC power systems may be required on activation of an actuation signal (without loss of power to the bus important to safety).

(ix) Paragraph 7.50 states: "Standby power sources should be independent of electrical power sources and power sources for instrumentation and control systems, other than those sources in their own division" [36]. However, it does not account for EIDs with multiple modules and with passive safety systems and shared facilities. In particular, when EIDs with multiple modules are provided with shared facilities, capacity may be provided for all modules in accordance with the safety case, and no event on any module may adversely affect the integrity of power supplies to other modules.

(e) Alternate AC power supplies (paras 8.1–8.18). These paragraphs offer guidance for the design of the electrical power supplies when neither off-site power sources nor on-site AC power sources are available (a station blackout event). However, they may not reflect the functional specifications, sizing and qualification requirements that are specific to EIDs.

4.3.7. IAEA Safety Standards Series Nos SSG-39, Design of Instrumentation and Control Systems for Nuclear Power Plants, and SSG-51, Human Factors Engineering in the Design of Nuclear Power Plants

4.3.7.1. Areas of applicability

Overall, the guidance provided in SSG-39 [37] and SSG-51 [38] can be applied to EIDs.

4.3.7.2. Areas of non-applicability

No areas of non-applicability were identified.

4.3.7.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) Life cycle models (paras 2.10–2.37 of SSG-39). The existing guidance in SSG-39 [37] remains essentially valid. However, it may not include the following considerations:
   (i) The notion of a multi-module plant, where the life cycle of the plant needs to be distinguished from the life cycle of a module, and the I&C of the plant needs to be distinguished from the I&C of a module, and where the following apply:
      — All the modules of a plant are usually of the same design (possibly with small differences), except when some, but not all, of the modules have been upgraded.
      — Some modules may still be in construction while others are already in operation.
      — Some modules may already be in decommissioning while others are still in operation.
      — There is a fleet of plants (with modules of the same design) supported by shared support centres, where the life cycle of the fleet needs to be distinguished from the life cycle of the plants.
   (ii) As mentioned above, a multi-module plant may have modules in different phases of a life cycle (e.g. installation/commissioning, operation, decommissioning). Multiple operational modules can be in different operational states (e.g. refuelling, outage, AOO, DBA). The impact on HSI design and whether SSG-39 provides sufficient coverage for such considerations may not be included.

(b) Activities common to all life cycle phases (paras 2.38–2.91 of SSG-39). The existing guidance in SSG-39 remains essentially valid. However, it may not include the following considerations:
   (i) Configuration management and documentation for multiple modules. In particular, it may be important for operation and maintenance, upgrades, lessons learned or big data analyses to rigorously identify what is common among modules and what is specific to each module in a plant or to a fleet of plants with modules of the same design.
   (ii) Hazard analysis in a multi-module context, where the same I&C hazard might affect multiple modules (of a plant or of a fleet) concurrently (in particular, hazards related to computer security).
   (iii) Verification and validation in a modular construction context: transportation of I&C modules built and verified in a factory could be explicitly defined as a phase in the I&C life cycle model.
   (iv) Probabilistic safety analysis and safety assessment could benefit from guidance on how multi-module aspects and sharing of systems may be considered.
(c) Digital systems. Paragraphs 7.66–7.147, 7.165–7.175, 7.148–7.164 and 9.1–9.103 of SSG-39 [37] provide specific recommendations on digital systems that remain essentially valid for EIDs. However, regarding computer security (paras 7.101–7.130), the existing guidance may not include the following considerations:
   (i) Computer security during transportation to site of modules of EIDs that are fully assembled, configured and tested in a factory;
   (ii) Possible computer security issues raised by staged construction;
   (iii) Possible computer security issues and benefits related to reliance on remote monitoring and support centres;
   (iv) Possible computer security issues and benefits related to the existence of multiple, quasi-identical modules in many different geographical sites;
   (v) Separation (from a standpoint of computer security) between the modules of a multi-module plant.
(d) DiD (paras 2.56, 2.81, 3.6–3.16, 4.3–4.40, 6.25, 6.71, 7.11, 7.25, 8.51 and 9.62 of SSG-39). The existing guidance in SSG-39 on DiD remains essentially valid. However, it may not cover the case of multi-module plants and the difference between the DiD of a module and the DiD of the plant. In particular, it does not address the circumstances in which it might be acceptable for several modules to share the same I&C systems.
(e) Identification of I&C functions (paras 3.1–3.6 of SSG-39) and allocation (paras 4.1–4.29 of SSG-39). The existing guidance in SSG-39 on the identification of I&C functions remains essentially valid. However, it does

not cover the case of multi-module plants where some functions are related to a module while others are related to the plant (in particular, in the case of shared plant systems).

(f) Range of operational and accident conditions (paras 3.14–3.16 of SSG-39). It is important for SSCs to meet all requirements when subjected to the range of expected operational, accidental and environmental conditions, many of which will depend on the design of the plant. While the range of operational and accident conditions for water cooled SMRs would be similar to those in existing WCRs, for other EIDs the conditions may be substantially different. Hence, the identification of the range of these conditions associated with plant states is not included for each specific EID.

(g) Environmental qualification (paras 6.96–6.107 of SSG-39). The existing guidance in SSG-39 on environmental qualification is generally valid for EIDs. The methods for establishing the environmental qualification of EID SSCs will broadly follow the process and methods established for large WCRs. However, the environmental profiles associated with operational states and accident conditions required for qualification will vary from one EID type to another. These paragraphs do not include the way in which I&C SSCs of EIDs meet all requirements when subjected to the expected range of environmental conditions.

(h) Operator manual actions (paras 3.13–3.15, 7.12, 7.13, 7.18–7.25 and 8.4–8.75 and annex III of SSG-39). The existing guidance in SSG-39 on manual actions remains essentially valid, but it does not consider possible co-activity operation, co-activity plant systems or both in multi-module plants.

(i) Internal and external hazards (paras 2.56–2.65, 6.30–6.37, 6.81 and 6.108–6.112 of SSG-39). The existing guidance in SSG-39 on internal and external hazards remains essentially valid, but the meaning of internal and external hazards may be different. For example, a hazard in a module or unit may become a hazard to other modules or units. In addition, there may be a distinction between plant-wide hazards (be they internal or external to the plant) and module level hazards (affecting only one module at a time, be they internal or external to the module).

(j) I&C system architecture (paras 4.1–4.40 of SSG-39). The existing guidance in SSG-39 [37] on the overall I&C architecture and individual I&C systems architecture remains essentially valid for EIDs. However, a distinction between the overall I&C architecture of the plant and the overall I&C architecture of individual modules is not considered in the case of multi-module plants with co-activity operation, co-activity plant systems or both.

(k) Classification of I&C systems (paras 5.1–5.13 of SSG-39). The existing guidance in SSG-39 [37] on classification remains essentially valid for EIDs. However, it does not consider that if an EID generates low power (even considering that an SMR may have multiple modules), relatively long grace times may be available before any manual action is required, and that the use of passive safety features and systems may have an impact on safety classification.

(l) General recommendations for all instrumentation and control systems important to safety (paras 6.6–6.76, 6.135–6.212 of SSG-39). The existing recommendations of SSG-39 remain essentially valid for EIDs. However, the following aspects are not covered:
   (i) For testing and testability in operation, accessibility to sensors could be problematic during operation in the case of integrated designs (SMRs).
   (ii) If the modules of a multi-module plant are served by a single I&C system, this may have consequences for control of access.

(m) Sensing devices and measurement techniques (paras 7.1–7.9 of SSG-39). The existing guidance in SSG-39 [37] regarding sensing devices remains essentially valid for EIDs. However, it does not address the issue of accessibility (e.g. for testing and calibration), which could be more difficult in the case of integrated designs (SMRs).

(n) Human–system interface (paras 8.47–8.93 of SSG-39). The general principles outlined in SSG-39 can still provide guidance for EIDs with respect to human factors engineering. However, the following additional considerations may not be included:
   (i) Shared control rooms. SSG-39 [37] does not provide specific guidance on shared control rooms. In particular, the following guidance may not be covered:
      — How situational awareness of all modules can be supported, particularly in the case of accident monitoring, when variables require continuous visibility.
      — Response or recovery of one module may include design features (visual or audible) that minimize the potential for the operator to neglect changes in other modules.
      — The extent to which monitoring and control of modules within a shared control room are to be integrated or separated within or between workstations and other HSI components, including equipment dedicated to specific operational states (e.g. maintenance, refuelling) or safety functions.
   (ii) The scalability of a shared control room. This includes how HSI technology, as well as visual and control space, can be designated, expanded or adapted to optimize control of added modules while reducing the possibility that operators may be confused when working with several modules. Limits to scalability may also be addressed

so that human performance is not negatively affected by physical or mental oversaturation. It is important that such changes align with human limitations (i.e. it may be appropriate to limit the number of reactor modules that can be monitored by one operator).

(iii) Related to scalability, the flexibility of a control room to support outages, refuelling, AOOs and emergencies. Addressing an affected module while avoiding impact on other operating modules may be a challenge. There may be consideration for dedicated management or response facilities that can accommodate any module.

(o) HSIs for common systems. Some EIDs, especially SMRs, may have multiple, separate modules and supporting systems but they are also likely to have common systems between modules. SSG-39 does not include the following considerations:

   (i) The interdependence, overlap or redundancy of these HSIs to minimize CCFs of HSIs that can affect multiple modules.

   (ii) When and how these HSIs for common systems are modified to include additional modules may be an issue for SMRs, or in general for EIDs. This may be a particular issue if there are modules in operation while new modules are being constructed.

   (iii) Guidance to ensure that adequate HSIs are provided to isolate modules from these common systems, where necessary, for maintenance and other activities.

(p) HSIs for management of life cycle phases and operational states. The HSIs for the simultaneous management of different stages in the lifetime and operational states of modules are not addressed in SSG-39 [37].

(q) Module to module differences. SSG-39 [37] and SSG-51 [38] do not address the potential for module to module differences, so they do not cover the possibility that these differences may be intentional or unintentional, making situational assessment and response planning more difficult. The differences between modules can lead to different HSIs, different procedures and different responses, thus increasing the potential for operator error.

4.3.8. IAEA Safety Standards Series No. SSG-62, Design of Auxiliary Systems and Supporting Systems for Nuclear Power Plants

4.3.8.1. Areas of applicability

Overall, guidance on the design of auxiliary systems provided in SSG-62 [39] can be largely applied to the design of auxiliary systems of EIDs.

4.3.8.2. *Areas of non-applicability*

The following areas of non-applicability were identified:

(a) The existing guidance in SSG-62 [39] related to the safety classification of some auxiliary systems may not be directly applicable to designs that operate with passive safety systems, which do not need the support of auxiliary systems in accident conditions. Consequently, the guidance related to the safety classification level of the heat transport system (paras 4.25–4.44), the compressed air system (paras 4.94–4.107), some HVAC systems (paras 4.108–4.170) and the supporting systems for the emergency power supply (paras 4.233–4.267) may not be applicable.
(b) In addition, regarding lighting and emergency lighting systems (paras 4.171–4.179), the list of locations where the emergency and station blackout lighting systems must be installed may not be fully applicable, since the design of some EIDs relies more on passive systems, which may not need manual actions during accident conditions.

The following recommendations in paras 4.268–4.289 of SSG-62 [39] concerning other systems may not be applicable:

(c) Equipment and floor drainage system. The reinjection of highly contaminated liquids from the auxiliary buildings or secondary containment into the reactor pressure vessel in accident conditions may not be applicable to some EIDs, such as LFRs, SFRs and HTGRs.
(d) Demineralized water reserve and associated system. Large WCRs may implement long term residual heat removal by the steam generators in the case of loss of the residual heat removal system. The recommendations may not be applicable to non-water cooled SMRs and other EIDs.

4.3.8.3. *Identified gaps and areas for additional consideration*

The areas not covered by the recommendations in paras 4.25–4.44 of SSG-62 [39] concerning heat transport systems are the following:

(a) The safety classification of the heat transport system may be assigned according to a graded approach, since safety systems may operate in passive mode in accident conditions, and consequently the active cooling of components may not be needed, and the diversification of the chilled water system may not be necessary if this system does not provide a function of safety category 1.

(b) Other functions may be added if the cooling of some SSCs is performed by a heat transport system, such as the cooling of the reactor cavity of an HTGR, SFR or LFR. On the other hand, for some EIDs, the cooling system for this cavity is operated in passive mode in accident conditions.

The recommendations in paras 4.45–4.72 of SSG-62 [39] concerning process and post-accident sampling systems may need to be revisited to consider specificities of EIDs (e.g. sampling of gas, sodium, lead), as follows:

(c) For HTGRs during normal operation, the helium sampling system is used to sample the primary helium coolant for analysis of moisture, chemical impurities and radioactivity. During accident conditions, the process and post-accident sampling system is used to sample gas in the reactor building.
(d) For LFRs and SFRs during normal operation, the process system provides sampling of the reactor coolant, cover gas and purification system. During accident conditions, the process sampling system provides sampling from the reactor containment to detect radioactive releases from the primary side.

Many of the recommendations in paras 4.73–4.93 of SSG-62 [39] concerning process radiation monitoring systems are applicable to EIDs. Monitoring the activity of the steam generator may not be applicable to some non-WCRs. However, where pressures are such that a leak in a steam generator tube results in water flowing into the primary coolant, such a leak may be detected by other means, such as monitoring of the moisture in the primary helium coolant for HTGRs, monitoring of the sodium–water reaction for SFRs and of the lead–water reaction for LFRs.

The recommendations in paras 4.94–4.107 of SSG-62 [39] concerning compressed air systems may not cover the following points:

(e) For EIDs using passive safety systems, the compressed air system may not be used as a supporting system of safety class 1 or 2. In this case, the safety classification level would be set using a graded approach.
(f) If a compressed air system is a shared system supporting a safety function in a multi-module EID, design provisions may be needed if an event in one module affects the ability of the system to support the other modules.

The recommendations in paras 4.108–4.170 of SSG-62 [39] concerning heating, ventilation and air-conditioning systems may not cover the following points:

(g) For large WCRs, dynamic confinement, created by the HVAC system, avoids contamination spread. SSG-62 [39] does not consider the possibility that this could be achieved by passive means. For instance, in the case of a station blackout, the design could create a natural airflow from a lower to a higher radioactive zone.
(h) For some light water SMRs, the integrity of the fuel building in case of boiling of the spent fuel pool may be provided by the HVAC system, which passively evacuates humid air and steam via the exhaust line of the ventilation. This configuration is not addressed by SSG-62, so two issues are raised:
   (i) The dynamic confinement principle required by SSR-2/1 (Rev. 1) [3] (mentioned in the previous point) may not be met by some EIDs.
   (ii) The efficiency of the iodine trap usually used is strongly dependent on the air humidity.
(i) Some ventilation systems of EIDs may have a safety classification level determined according to a graded approach, especially when the ventilation is not needed to reach and maintain a safe shutdown state.
(j) Buildings may be shared between different modules in a multi-module EID, so it is important that the ventilation system is designed to operate during an accident in one module without affecting the capability to reach a safe state in the remaining modules.
(k) For HTGRs, a primary break would cause a quick increase of the pressure of the reactor building owing to the release of high pressure primary coolant. To maintain the integrity of the reactor building, a specific device (usually calibrated valves are used) needs to limit the pressure below the design pressure of the reactor building. When the pressure in the reactor building has sufficiently decreased, the activation of the pressure limiting device may not be necessary.
(l) One of the objectives of the MCR HVAC is that plant operators are adequately protected against the effects of accidental releases of toxic or radioactive gases. This function may be ensured on an EID by a passive system, but this possibility is not addressed by SSG-62 [39].

The recommendations in paras 4.180–4.198 of SSG-62 [39] concerning overhead lifting equipment may not cover the following points:

(m) The use of lifting equipment other than overhead lifting equipment.

(n) The use of neutral gas in the removal of components from the primary and secondary circuits of SFRs and LFRs.
(o) Lifting equipment and practices for transportable reactor modules in multi-module and single module configurations.
(p) The recommendations in paras 4.199–4.232 of SSG-62 [39] concerning systems for the treatment and control of radioactive waste and radioactive effluents may not cover the following points for three types of system:
    (i) Systems for the treatment of gaseous effluents. For non-water cooled EIDs, the treatment of gaseous effluents is similar to that in WCRs but the measures for managing polonium in LFRs are not included.
    (ii) Systems for the treatment of liquid effluents. No specific areas that are not covered were identified.
    (iii) Systems for the treatment of solid waste. The recommendations do not recognize the possibility that the solid waste of several EIDs, located at different sites, could be treated by a common waste management installation.

Most of the functions and recommendations in paras 4.233–4.267 of SSG-62 [39] concerning supporting systems for the emergency power supply and the alternate power supply are generally valid for EIDs. However, the recommendations may not cover the following points:

(q) The safety classification and the engineering design rules applicable to the supporting systems of the emergency power source. Some EIDs rely on passive design features to achieve the safety functions required to reach a safe shutdown state in accident conditions. The safety functions performed by the safety systems required for safe shutdown may depend on safety class 1 DC power sources, such as batteries. The AC emergency power supply may be needed only for recharging the batteries, so a DC power supply may be used to support restoring the inventory to passive safety systems, and in this way maintain core cooling and a safe state for the overall plant for an extended duration. Accordingly, the DC power systems may have to be provided with sufficient capacity to maintain the plant in a controlled state until off-site power supplies can be restored, or alternative on-site AC supplies can be provided. Consequently, the safety class of the emergency on-site AC power system of an EID design with passive systems may be assigned according to a graded approach. The AC emergency power source may then be classified according to its safety significance and satisfy corresponding applicable design rules.
(r) Fuel reserve. If an EID is designed with black start capability (can start up from a completely de-energized state without receiving energy from the

grid), the fuel oil storage, air start, DC power and other supporting systems must consider this operation mode.

(s) Shared supporting systems. Paragraph 4.236 states that "each emergency power source should be provided with its own completely independent supporting systems" [39]. This recommendation may not cover the possibility that SMRs, or in general EIDs, may have shared systems to support AC emergency power sources.

(t) The functional specificities of the alternate power source of EIDs.

The recommendations in paras 4.268–4.289 of SSG-62 [39] concerning other systems may not cover the following points:

(u) Equipment and floor drainage system:
    (i) For SFRs, the drainage of components removed from the sodium circuit or the sodium circuit as a whole;
    (ii) Collection of the gas effluents coming from the gas cover used by SFRs or LFRs.

(v) Demineralized water reserve and associated system. Large WCRs may implement long term residual heat removal by the steam generators in case of loss of the residual heat removal system. On the other hand, the residual heat removal in light water SMRs may be performed by a system operating in natural circulation that transfers the heat directly to a water reserve. The reserve is consumed by evaporation and, consequently, it may have to be resupplied in some cases. However, the recommendations do not include the resupply of this reserve in the long term.

(w) In addition, other functions must be achieved by auxiliary systems for non-water cooled EIDs that may not be covered by the current recommendations. Some examples are the following:
    (i) Specific HTGR systems. Examples are the helium purification system that removes chemical impurities of the helium coolant, as well as the graphite dust and radionuclides in some designs; also, the helium storage and supply system that maintains the pressure of the primary helium coolant.
    (ii) Specific LFR and SFR systems. Examples are systems for cover gas storage, for detection and removal of impurities in the cover gas and the coolant system, and for coolant heating.

(x) There are potential supporting systems that are not covered by SSG-62 [39]. For example, heating systems and their controls may be provided in non-water cooled EIDs for the primary coolant to prevent loss of this coolant circulation by coolant freezing. They are designed with the objective that the temperature distribution and rate of change of temperature are maintained within prescribed limits. Section 3 of this publication mentions other novel supporting systems.

Finally, the guidance in SSG-62 [39] may not consider the implications of an EID with several modules, so the following aspects may not be covered:

(y) A support or auxiliary system may be shared among several modules in an EID (such as an SMR) or among units in a single site, and guidance may not be provided in relation to module to module interactions so that, for example, an accident on one module does not prevent normal shutdown of the others.
(z) Guidance for avoiding unwanted interactions in situations such as: (i) the construction, installation etc. of a new module alongside other modules that are already in operation, and (ii) the connection of a new module to a shared auxiliary system that is already in use by other modules.
(aa) A multi-module EID may extend its power capacity during its lifetime through additional module installation. Where auxiliary systems are shared, the initial design may anticipate the requirements of the additional modules; the standard may not discuss this topic.

4.3.9. IAEA Safety Standards Series No. SSG-63, Design of Fuel Handling and Storage Systems for Nuclear Power Plants

4.3.9.1. Areas of applicability

The guidance provided in SSG-63 [40] has broad applicability to water cooled EIDs. However, its applicability to new types of fuel and associated fuel handling and storage system is not straightforward. Sections 4.3.9.2 and 4.3.9.3 present specific areas of non-applicability and gaps.

4.3.9.2. Areas of non-applicability

The following areas of non-applicability were identified:

(a) Paragraph 2.19 (c) of SSG-63 [40] is not applicable to pebble bed HTGRs, because this design uses hundreds of thousands of randomly scattered spherical fuel elements, and it is impossible to identify each one.
(b) Paragraphs 3.1–3.140 of SSG-63 apply mainly to wet storage of spent fuel, so the items important to safety include the pool structure, pool liner, pool cooling systems, pool purification systems, make-up systems, gates and fuel storage racks, which are not applicable to HTGRs.

(c) Paragraphs 4.1–4.58 of SSG-63, which concern fuel handling equipment, loads, hypothetical events and refuelling operations, are not generally applicable to HTGRs.
(d) Paragraphs 5.1–5.21 of SSG-63 are not generally applicable to the fuel handling and storage systems of HTGRs. Pebble bed HTGRs use spherical fuel elements, so there is no fuel disassembly and reassembly. Other components mentioned (e.g. reusable reactor items) are also not relevant to pebble bed HTGRs. In the case of HTGRs with prismatic fuel and replaceable reflector blocks, the applicability of these paragraphs is better, as the handling involves similar systems designed for WCR irradiated fuel.

4.3.9.3. Identified gaps and areas for additional consideration

SSG-63 [40] focuses on the handling and storage of fuel, as it pertains to reactor operations. The wet or dry storage of SNF is covered by SSG-15 (Rev. 1) [80]. SSG-63 does not extend to the specifics of non-WCRs and advanced fuel types. Specific examples of gaps are provided below:

(a) The following aspects of the fuel handling and storage system of a pebble bed HTGR (para. 1.6) are not covered: burnup measurement of irradiated fuel elements; on-line refuelling of fuel elements that have not reached the target burnup; specifics of fuel element identification regarding inspection, transportation and storage; detection, separation and storage of damaged fuel elements; and dry storage of fresh fuel and spent fuel.
(b) Liquid metal cooled fast reactors (SFRs and LFRs) may have high enrichment and, thus, great amounts of decay heat needing storage of the spent fuel in liquid metal, which may introduce additional challenges. In general, such considerations are associated with the cooling, shielding, monitoring, capacity and other factors of storage systems, such as dry storage of fresh and spent fuel in HTGRs and storage under liquid sodium for SFRs (including ex-vessel storage and in-vessel storage of fuel and considerations for cooling)[26] (section 3 of SSG-63).
(c) Considerations associated to the fuel storage for HALEU are not covered.
(d) The on-power refuelling, as well as the associated considerations to ensure the integrity of the pressure boundary and the fuel cooling functions during loading and unloading in pebble bed HTGRs, are not covered in section 4 of SSG-63. There may be also potential new failure modes such as blocking and bridging of fuel elements. The design of the fuel handling system also considers fuel handling under normal operation conditions to ensure

[26] Similar considerations may be relevant for LFRs.

continuous operation of the HTGR plant and transport of fuel elements by force or gravity. Furthermore, new design features may be needed to deal with atmosphere isolation during the on-line refuelling process to prevent air from entering the loop and the radioactive gas from escaping. In addition, the fuel elements in the system are counted to facilitate the management of the fuel elements in and out of the core; a specific device may be installed in the system to separate out and store damaged fuel elements. Devices for measuring the burnup of the fuel elements may also be installed, as well as a means of calibrating the burnup measurement.

(e) The special features related to the handling of equipment in SFR vessels, such as equipment that serves as a reactor cover gas boundary during normal power operation, are not covered by section 4 of SSG-63. Fuel misloading, fuel assembly drop during handling and cooling failure caused by the fuel assembly becoming stuck during transportation may not be covered. Cleaning of sodium during fuel transfer from a sodium filled pool to a water pool may also not be addressed. Similar points may also be made in relation to LFRs.

(f) Another key gap is that factory-sealed lifetime cores[27] are not addressed in SSG-63. These have implications on safety, security and safeguards. Examples include the following:
    (i) Paragraph 2.15 on security does not address the concept of lifetime cores. Provisions on security considerations for initial fuelling, operation and end of life core for lifetime cores are not considered.
    (ii) Paragraph 2.16 on safeguards does not cover specialized fuels and the potential for lifetime cores. Safeguards considerations for novel fuels (e.g. fuels for HTGRs) are not covered. These fuels are also not covered in relation to the stages of initial fuelling, operation and end of life for lifetime cores.

4.3.10. IAEA Safety Standards Series No. SSG-64, Protection Against Internal Hazards in the Design of Nuclear Power Plants

4.3.10.1. Areas of applicability

Section 2 and appendix I of SSG-64 [41] can be readily applied to SFRs, LFRs and HTGRs. Sections 3, 4 and appendix II of SSG-64 can generally be applied, with a degree of interpretation, to non-WCR types (see Section 4.3.10.3).

---

[27] A lifetime core may be defined as a core with fuel components installed in the fabrication phase, so fuel will not be extracted from the core during operation. Therefore, no fuel handling is required during operation.

4.3.10.2. Areas of non-applicability

No areas of non-applicability were identified.

4.3.10.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) Paragraph 3.19 of SSG-64 [41] does not provide examples specific to EIDs and to possible industrial applications that may be linked to EIDs.
(b) Paragraph 3.32 addresses the interactions between the units at a multi-unit site. However, the recommendations do not account for the specifics of some EIDs, such as multi-module aspects.
(c) Paragraph 4.7 lists specific aspects to consider in the design in order to minimize the fire hazard. However, this list does not account for specific materials and resulting fire hazards introduced by EIDs (e.g. coolants, advanced materials used in the core — such as graphite in HTGRs, assuming that graphite fire is not precluded by using high quality nuclear grade graphite).
(d) Paragraph 4.12 discusses systems containing flammable liquids and gases and stresses the high importance of the integrity of such systems, considering corrosion or other destructive effects. However, specific considerations on the structural integrity of reactor coolant systems of EIDs with novel coolants, such as sodium, are not included. These recommendations on design do not cover the systems providing storage and treatment of novel coolants or the storage of specific types of nuclear fuel.
(e) Paragraphs 4.15–4.17 provide additional recommendations on the control of potential ignition sources and on minimizing the introduction of additional sources via the design approach, thus making the SSCs safe. These recommendations are not specific to systems storing or treating novel coolants or to systems storing fuel.
(f) Paragraph 4.28 provides recommendations to prevent spurious activation of fire mitigation systems. However, it does not include specificities related to mitigation of fires of metallic coolants or advanced fuel structural materials.
(g) Paragraph 4.36 provides a general list of objectives for mitigation of the effects of fire. This list does not capture the specifics of fires of metallic coolants, notably spray fires, or of advanced fuel structural materials.
(h) Paragraphs 4.45–4.47 provide recommendations specific to internal fires that may result in the release of radioactive substances, such as the housing of systems that can release radioactive material in the case of fire in fire

compartments, and heat and smoke venting. These recommendations do not include fires associated with metallic coolants.

(i) The recommendations provided in paras 4.50–4.59 are very specific to current WCRs, so they may not cover the specifics of technology used in EIDs.

(j) Paragraphs 4.61 and 4.62 provide general recommendations on explosion hazard identification and analysis. However, the list of explosion hazards may not include the new materials introduced by non-water cooled EIDs; for example, when chemical explosions are considered.

(k) Paragraphs 4.63–4.72 recommend general measures to be taken in the design in order to minimize the likelihood of explosions. The recommendations focus on the storage of flammable gases and liquids, as well as on minimizing the generation of specific flammable substances. However, they do not cover additional sources of flammable or explosive substances that may be used or generated at non-water cooled EIDs owing to factors such as operation, introduction of new materials and industrial applications.

(l) Paragraphs 4.73–4.77 provide recommendations on how explosion hazards could be mitigated by specific design measures. However, additional explosion hazards identified for non-water cooled EIDs that are due to the use of new materials or industrial applications may not be covered.

(m) Paragraphs 4.142–4.153 provide general recommendations on the identification and analysis of internal flooding hazards. However, spills and floods from RCS coolants in non-water cooled EIDs are not covered.

(n) Paragraphs 4.154–4.160 offer recommendations on the prevention of internal flooding hazards by design. However, para. 4.160, although relevant for fire extinguishing systems for any reactor technology, does not include examples related to EIDs.

(o) Paragraphs 4.161–4.165 address recommendations to be considered in the design of WCRs to mitigate internal flooding and its effects. However, floods from the coolants of non-water cooled EIDs, their effects and their mitigation are not included.

(p) The examples of events or categories of heavy load drop provided in paras 4.173–4.175 reflect the design and operation of land based, stationary WCRs. Hence, the recommendations and examples provided may not include some categories of heavy load drop events in EIDs that are not considered in the former plants.

(q) Paragraphs 4.200–4.204 provide general recommendations on the identification and analysis of hazards from releases of hazardous substances inside the plant. However, new materials and chemical substances may be used in the operation or may be generated during accident conditions in

EIDs. Therefore, these recommendations may not account for the specifics of these materials and substances.

(r) Paragraphs II.53–II.58 provide recommendations on the design of water based extinguishing systems, and they are generally applicable to EIDs. However, these recommendations do not cover the prevention of potential interaction of water with specific materials, such as sodium, introduced by some EIDs.

(s) Paragraphs II.75–II.81 provide recommendations for the design of gaseous extinguishing systems. Since such systems are usually used for extinguishing fires in electrical equipment, these recommendations are considered generally applicable to EIDs. However, they do not include considerations for the selection of appropriate extinguishing gases when there are concerns about chemical reactions with specific materials introduced by EIDs.

(t) Paragraphs II.82–II.87 offer recommendations for the design of dry powder and chemical extinguishing systems. Since such systems are usually used for extinguishing fires of flammable liquids or electrical equipment, these recommendations are considered generally applicable to EIDs. However, they do not cover considerations for the selection of appropriate extinguishing substances when there are concerns about chemical reactions with specific materials introduced by EIDs.

(u) Paragraphs II.88–II.92 provide recommendations for portable and mobile fire extinguishing equipment. However, these recommendations do not include considerations for the selection of appropriate extinguishing substances when there are concerns about chemical reactions with specific materials introduced by EIDs.

4.3.11. IAEA Safety Standards Series No. SSG-68, External Events Excluding Earthquakes in the Design of Nuclear Power Plants

4.3.11.1. Areas of applicability

Since the external events considered in SSG-68 [42] affect mainly the external envelopes of a facility (e.g. buildings), the guidance is not linked to any particular reactor technology. Therefore, there is no limitation in the scope of SSG-68 in this regard.

4.3.11.2. Areas of non-applicability

No areas of non-applicability were identified.

4.3.11.3. Identified gaps and areas for additional consideration

TNPPs, including marine based reactors, are not covered if they are not placed at a specific site.

4.3.12. IAEA Safety Standards Series No. SSG-67, Seismic Design and Qualification for Nuclear Power Plant

4.3.12.1. Areas of applicability

There is no explicit limitation in the scope of SSG-67 [43] to any particular reactor technology.

4.3.12.2. Areas of non-applicability

The following areas of non-applicability were identified:

(a) Seismic design as considered in SSG-67 [43] is applicable only to land based facilities.
(b) Floating reactors are not directly subjected to seismic waves, since they are seismically isolated from the ground by the water. However, when moored at a particular location, seismic design may apply to appurtenances, protective structures or nearby slopes, whose seismic failure might affect the safety of the reactor. SSG-67 covers these aspects.

4.3.12.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) TNPPs, including marine based reactors, are not covered if not placed at a specific site.
(b) While SSG-67 [43] has no explicit limitation in its scope to a particular reactor technology, it is clear that it was written with WCR technology in mind and, consequently, guidance for some technology specific SSCs present in other technologies might be absent.

4.3.13. IAEA Safety Standards Series No. SSG-69, Equipment Qualification for Nuclear Installations

4.3.13.1. Areas of applicability

The recommendations provided in SSG-69 [44] regarding a structured approach to the establishment and preservation of equipment qualification in nuclear installations remain essentially valid for EIDs.

4.3.13.2. Areas of non-applicability

No areas of non-applicability were identified.

4.3.13.3. Identified gaps and areas for additional consideration

Additional considerations may be necessary for the qualification of novel materials, manufacturing methods and SSCs, as well as sensors, instrumentation cables and connections, and their associated penetrations so they can operate during anticipated service conditions in all plant states (e.g. operational states and accident conditions), especially for non-WCRs technologies.

4.3.14. IAEA Safety Standards Series No. NS-G.1.13, Radiation Protection Aspects of Design for Nuclear Power Plants

NS-G-1.13 [45] is currently under revision. Most of the identified gaps will be resolved in the revised version, since new recommendations (including new areas and topics, such as modern technologies, lessons learned from accidents and operating experience) are introduced in a technology neutral manner in this revision.

4.3.14.1. Areas of applicability

NS-G-1.13 is generally applicable to all reactor types. There are some gaps as discussed below.

4.3.14.2. Areas of non-applicability

No areas of non-applicability were identified.

4.3.14.3. Identified gaps and areas for additional consideration

NS-G-1.13 does not cover the following top level aspects (although the revised version will resolve them):

(a) Some EIDs are likely to have significant radiation exposure disparities associated with different operational states and periods. The use of features such as compact layouts/arrangements and inaccessible areas within the NPP, with long intervals between maintenance and/or in-service inspections, can lead to a dose profile of low doses (individual and collective) during uninterrupted operational periods of 5 to 10 years, followed by short periods of much higher exposures during major component replacements (such as a reactor vessel). However, dose targets for distinct operational periods, as well as their consideration early in the design process for a reactor concept as a long term practice, may not be covered.
(b) Average collective dose considerations may be used by EID developers to make a case to not further optimize exposures as part of the design process, particularly in instances where individual exposures may approach dose limits. These considerations may be particularly important for first of a kind deployments, where the amount of operating experience to support planning for ageing management, maintenance and outage management may be low. However, best engineering practices and setting dose targets in the absence of operating experience for new designs, particularly for high risk activities, such as replacing major components, is not considered.
(c) Experience in the operation and maintenance of robotic and remote systems has shown that significant design planning and the need for special skills by plant staff needs to be considered from the outset to address radiation protection practices. Using safety analysis tools, such as hazard analysis, may be useful to identify potential events that could lead to unplanned maintenance under challenging circumstances, but they are not mentioned.
(d) Paragraph 1.9 of NS-G-1.13 [45] may not address non-water cooled EIDs.
(e) Paragraphs 2.7–2.9 may not include the establishment of dose targets for distinct operational periods (e.g. extended normal operations periods, component replacement periods).
(f) Regarding paras 3.1–3.3: (i) para. 3.1 may not include sources unique to EIDs (e.g. off-gas components); (ii) para. 3.3 may not identify types of EIDs; and (iii) annex III may also not cover related aspects.
(g) Paragraphs 3.10–3.13 may not include engineering practices that could be considered in the absence of operating experience for EIDs (particularly for high risk activities, such as major component replacements).

(h) Parts of paras 3.14–3.17 and figure 1 of NS-G-1.13 [45] address experience at relevant plants. However, there may not be relevant experience for EIDs. Accordingly, these paragraphs may not include establishing dose targets for distinct operational periods (e.g. extended normal operations periods and component replacement periods). In addition, they may not address guidance on setting targets for EIDs.
(i) Paragraphs 4.4–4.8 do not include guidance on the control of corrosion products for EIDs.
(j) Paragraph 4.9 does not include guidance on the control of fission products for EIDs.
(k) Parts of Section 5 are written for WCRs and may not cover other reactor types. These include paras 5.18–5.26, 5.79–5.82 and 5.99–5.102. In addition, paras 5.103–5.108 are technology neutral, but their scope may not include all technologies if control of gaseous emissions is more challenging than in existing commercial WCRs, heavy water reactors and gas cooled reactors. Furthermore, other paragraphs contain limited examples of operating experience with NPPs in Member States, which usually refer to PWR or BWR technology, and they do not include a wider range of illustrative examples to make them explicitly relevant to other technologies. These include paras 5.12–5.17, 5.34–5.37, 5.57–5.67, 5.68–5.78 and 5.92–5.96.
(l) The recommendations in the revised version of NS-G-1.13 will cover the effects of multi-unit sites during accidents. While these are mostly generic, they may not be relevant to some issues that are specific to EIDs. Examples include multi-module plants (such as SMRs) with potential size reduction of the EPZ, the possible increase in off-site population density, and more specific treatment of multi-modularity in a single plant. In addition, the recommendations do not include the specific characteristics of EIDs, such as power levels and inventories of radiological and non-radiological contaminants.
(m) Regarding paras 6.15–6.18, the possibility of introducing new radiation sources during decommissioning (as in EIDs) is not considered.

The following gaps were identified in section 7 of NS-G-1.13 [45]:

(n) The removal of big parts of an EID from the site and their dismantling in another location may reduce the sources of radiation at the site. This aspect at an EID site is not covered.
(o) The transportability of parts of the NPP, or entire modules (with or without fuel), from an NPP site to the demolition site, and the handling of those parts or entire modules of the NPP are also not covered.

(p) Locations of the radiation monitoring system that are specific to EIDs are not considered in paras 7.11–7.16.
(q) Locations or systems for effluent monitoring that are specific to EIDs are not included in paras 7.17–7.19.

The following gaps were identified in section 8 of NS-G-1.13:

(r) The specific characteristics of reactor technologies (e.g. neutron spectra, coolant materials in the primary and secondary circuits, and moderator materials) are not included.
(s) Aspects related to the transportation of the reactor for TNPPs are not covered.
(t) Paragraphs 8.1–8.7 include some guidance on the radiation monitoring of specific systems for WCRs. However, radiation monitoring of process systems that are specific to EIDs are not considered.
(u) A graded approach in the design of radiation monitoring systems is not mentioned.
(v) The listing of auxiliary facilities in paras 9.1, 9.2 is likely sufficient to support operations for EIDs. However, additional facilities (e.g. hot or warm cells) for EIDs may not be covered.
(w) Annex I (paras I–1 to I–7) does not cover the potential for significant disparities in exposure during different operational periods (e.g. extended periods of normal operation with low exposure and shorter periods of higher exposures during major component replacements).
(x) Annex II (paras II–1 to II–7) addresses sources of radiation in various systems for some reactor types (such as BWR, PWR, advanced gas cooled reactor and fast breeder with liquid sodium coolant), but it does not address source term information for EIDs. In addition, it does not cover TNPPs and possibly microreactors.
(y) Annexes III (paras III–1 to III–46) and IV (paras IV–1 to IV–17) consider source terms under accident conditions for some reactor types and accident scenarios, such as steam line breaks, SGTRs and fuel handling accidents. However, they do not include EIDs and their postulated accidents.

4.4. CONSTRUCTION

4.4.1. IAEA Safety Standards Series No. SSG-38, Construction of Nuclear Installations

4.4.1.1. Areas of applicability

The guidance provided in SSG-38 [46] is at a high level and is not specific to any technology so that it has broad applicability to both WCRs and EIDs. However, there are some gaps and areas for additional consideration, which are discussed below.

4.4.1.2. Areas of non-applicability

No areas of non-applicability were identified.

4.4.1.3. Identified gaps and areas for additional consideration

The following considerations are not currently addressed by SSG-38 [46], because they represent new approaches and practices emerging with EIDs.

The use of advanced manufacturing and construction methods and novel and recycled materials:

(a) Currently, many advanced manufacturing and construction methods and novel materials are being considered (see Section 3.3) that are not fully qualified, do not have existing consensus codes and standards, or both. Industry codes and standards for novel manufacturing, construction, inspection and test methods may not yet be proven to be suitable for nuclear manufacturing and construction or may not exist. This does not necessarily indicate a gap in SSG-38.

(b) Some advanced manufacturing techniques may limit the ability to fully verify the control of processes via subsequent (post-build) inspection and testing. For these cases, the manufacturer's identification, qualification and control of the processes is critical in ensuring the integrity of safety related items and proper disposition of non-conformances or in-service inspection results. Paragraphs 5.17 and 5.18 of SSG-38 are broad but may not address these unique aspects.

The incorporation of below grade structures:

(c) Consensus standards for below grade structures represent a gap in industry codes and standards, but not a gap within SSG-38, which is a high level publication. The use of below grade nuclear structures may impose the need to develop health monitoring techniques different from the conventional construction and inspection methodologies that are typically used for above grade structures. Below grade structures exhibit unique consequences in the case of damage and therefore have stricter monitoring and mitigation requirements. In addition, geotechnical construction technologies and geotechnical engineering activities are more important when using below grade structures. Detailed assessment of the soil characteristics as well as the surrounding rocks is needed to ensure long term reliability of these structures.

Reliance on factory based serial modular manufacturing and construction:

(d) In a factory setting, manufacturing can replace traditional field construction activities. SSG-38 provides no discussion of modular construction or prefabricated structural and mechanical modules. In particular, the smaller size (compared with conventional large WCRs) of some EIDs and SMRs offers the ability to increase factory fabrication and modularity. Shifting of construction activities to supplier facilities requires special attention as to the following points:
    (i) There is no specific discussion on the use of assessments of manufacturability, constructability combined with maintainability, operability and decommissioning during the design process, including consideration of modularity issues. Furthermore, serial factory construction of modular designs may be difficult because of differences in local design codes or individual customer requirements. Paragraph 4.16 does not include a specific discussion on manufacturability and constructability assessments and how they need to interface with operability and maintainability assessments and design activities. The roles of owners, contractors and field engineers in making improvements that result in constructible designs that do not impede safe operation and maintenance over the lifetime of an NPP are also important but not covered.
    (ii) Paragraphs 4.56 and 4.57 do not include discussion about modular construction and prefabricated structural and mechanical modules. Processes to assess any potential interferences, tolerances and fit-ups of modules between multiple locations and entities are not covered.

Furthermore, serial production may provide the opportunity to implement sampling for quality control.

(iii) The guidance in Section 5 about manufacturing and assembly is comprehensive, but it does not include examples or case studies involving complex modular systems or equipment or ties to experience databases.

(iv) There is no guidance aimed at ensuring that tests are conducted at the most appropriate stage of construction when items are accessible. Any testing of a complex modular system at the factory of origin will need to be factored into the deployment site's programme for acceptance for installation and commissioning. Asset management of items that are stored in preparation for shipment is also not covered.

(v) The complexity of factory made modules may introduce new receipt issues at a site. Guidance may not be provided on the evidence to be presented to the licensee to demonstrate that adequate levels of quality have been achieved.

(vi) There is no discussion on how factory built modules will be maintained, examined, inspected and tested once they are installed and connected. In some cases, supplementary inspections and tests may be necessary to verify fitness for service after receipt and installation.

(vii) The complexity of the supply chain and the shift of construction activities off-site could pose challenges for the licensee in terms of the manufacturers' safety culture and quality control to ensure that manufacture is in line with safety classification and design requirements. This may require a careful assessment of the supply chain, including a risk assessment, before the start of the construction. These and other important issues related to the supply chain, including pandemic, extreme weather conditions, political unrest and cyber security attacks, as well as proper plans and procedures that need to be in place before construction, are not considered.

Phased approach to construction:

(e) Some EIDs may adopt a phased approach to the construction of modules. They may also be built in proximity to an existing operating plant. There is no consideration of the impact of the potential co-location of a construction site around other sites (existing or not) and how this would be licensed — that is, whether in a phased way to extend the area of a licensed site or if a separate licence would be required per reactor site.

Collection and dissemination of information on construction experience:

(f) Section 4 of SSG-38 [46] addresses the collection and dissemination of information on construction experience. However, SSG-38 does not fully consider how the planning and conduct of construction activities may benefit by drawing from global experience and good practices. In the last decade, work has been done to establish construction experience databases to collect and learn from experience. It would be useful for any construction project to have, within its integrated management system, mechanisms to analyse construction experience and apply it to the planning and execution of activities.

Control of design information:

(g) Configuration management as well as managing field changes are key contributors to maintaining safety and meeting construction schedules. Technologies such as configuration management information systems, building information modelling, model based systems engineering and digital twins may be deployed before, during and after construction to facilitate control, storage and management of design information. Establishing a common platform that is easily accessible by all entities assists in providing the project management function with better control of design, construction, installation and commissioning activities.

Use of digital systems:

(h) There is no discussion of the use of 3-D modelling techniques (as an update of 2-D 'paper' drawings) in conformity control (e.g. to prevent interferences) or of new technologies such as model based systems engineering, digital twins and laser scanning to assist construction inspections.

Balance of plant:

(i) Proposals to functionally separate the balance of plant from the nuclear island will change how the balance of plant interfaces with the nuclear island in the safety analysis for the overall facility. For example, energetic events from the balance of plant may be 'buffered' from the nuclear island using intermediate heat transport loops, large heat capacitors[28] or both. This method generally means that the nuclear island would rely less on the balance of plant systems to manage events in the nuclear island. An example would be the traditional use of feedwater systems as a cooling loop. These considerations are related to the construction; for example, there is a potential need for physical and 'soft' boundaries to separate the different scope or quality needs during construction. However, these needs or best practices are still in development. These factors and relevant guidance, such as safety classification or requirements for intermediate loops, are important, but SSG-38 does not discuss them.

4.5. COMMISSIONING AND OPERATION

4.5.1. IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation

4.5.1.1. Areas of applicability

SSR-2/2 (Rev. 1) [19] is generally applicable to EIDs. Some gaps were identified and are described below.

4.5.1.2. Areas of non-applicability

No areas of non-applicability were identified.

4.5.1.3. Identified gaps and areas for additional consideration

The following potential gaps have been identified in SSR-2/2 (Rev. 1) [19] regarding operation and commissioning requirements for EIDs:

(a) Requirement 4: Staffing of the operating organization. Paragraph 3.12 implies that shift personnel will be located at the site. As some EIDs may be operated remotely, there are situations where a very limited number of personnel are present at the site, but this is not considered.

[28] A heat capacitor is an energy storage device such as a large tank of molten salt. Under normal operation, the nuclear island transfers heat to this system (with backup heat rejection paths available) so that the balance of plant can draw energy on demand, without the need for the reactor to load-follow.

(b) Requirement 7: Qualification and training of personnel. Some specific features are not considered, such as the training programme for remote operations, monitoring and control strategies for remote operations, multi-unit and multi-design site operations, and novel refuelling strategies.
(c) Requirement 8: Performance of safety related activities. There is a gap regarding the applicability of safety related activities for multi-unit EIDs and EIDs with alternative operating models including autonomous systems and remote monitoring and intervention capabilities.
(d) Requirement 10: Control of plant configuration. Configuration management for multi-reactor module plants — particularly for a single NPP with multi-reactor modules that may be located in shared structures — is not considered.
(e) Requirement 13: Equipment qualification. Some specific features are not included, such as the impact on the equipment qualification programme of higher temperatures, different chemistry, components integrated with the reactor and the importance of incorporating any lessons learnt from new operating conditions that influence equipment reliability.
(f) Requirements 14 and 16: Ageing management and programme for long term operation. This requirement does not consider lessons learned from new operating conditions and activities that may be relevant to EIDs and that can influence ageing management programmes and long term operations; for example, the influence of higher operating temperatures and of different chemistry and component materials. New activities may include replacement of major modules, including in some cases the whole reactor, and maintenance/refuelling that is performed off-site. Remote surveillance, inspection and testing techniques may also have an impact on ageing management.
(g) Requirement 20: Radiation protection. This requirement does not consider the need to ensure that applicable lessons learned from operations with novel coolants, maintenance practices and long refuelling intervals from EIDs are incorporated into radiation protection programmes.
(h) Requirement 22: Fire safety. This requirement does not include hazard assessments for the types of reactor coolant and the types of combustible material that may be present at EIDs. In addition, firefighting arrangements have to ensure clear coordination and cooperation for firefighting at multi-unit sites and where systems are shared.
(i) Requirement 25: Commissioning. There is a gap regarding commissioning at the factory versus at the NPP site and the role of the operator in both; this is especially important for offshore manufacturing, where standardized systems are being proposed. This may require, for example, on-site commissioning tests to check that the results obtained off-site are valid and nuclear safety checks that are additional to criticality testing at the manufacturer's site.

In particular, Requirement 25 does not include criticality testing at the manufacturer's site prior to transportation and the need for any additional on-site commissioning tests to verify that the results obtained off-site are still valid at the plant. In addition, the representativeness of commissioning tests has to be demonstrated to prove that passive safety systems will perform as expected in accident conditions.

(j) Requirement 27: Operation control rooms and control equipment. A gap in SSR-2/2 (Rev. 1) [19] exists for cases where NPPs may employ combinations of in-plant autonomous systems and local operator control facilities combined with the capability to monitor and intervene from a remote location.

(k) With respect to the habitability of control rooms (para. 7.7), the guidance might not include considerations from the perspective of operation being fully in-plant, partially in-plant and partially remote, or fully remote with minimum number of operators present at the reactor site.
    (i) In the case of multiple units or sites of EIDs, there may be a main control room that receives data from a series of localized control rooms, which in turn may obtain data from other control room(s) at the site(s). Guidance is not provided on establishing a control hierarchy or on coordinating operations from remote facilities.
    (ii) The type and function of supplementary control rooms also needs further consideration.

(l) Requirement 29: Chemistry programme. This requirement does not consider the chemistry programme consideration of the types of coolant used in non-water cooled EIDs and their impact on plant component materials.

(m) Requirement 30: Core management and fuel handling. The requirement is fully applicable to the operation of EIDs with solid fuels. However, it does not cover: (i) fuel that has no cladding (para. 7.24) and fuel defects (para. 7.25) in some EIDs; (ii) situations where different organizations are responsible for fuel loading and unloading if this takes place at the factory or at the site; and (iii) the electrical grid integration and in particular the resilience of the design to grid imposed short term shutdowns and restarts and the ability to load-follow.

(n) Requirement 31: Maintenance, testing, surveillance and inspection programmes. The use of remote monitoring for testing, inspection, maintenance and control is not currently considered. In addition, the maintenance, surveillance and periodic testing of passive equipment are not fully considered, particularly the issue of how the periodic tests can prove that the expected performances of passive systems are maintained throughout the plant lifetime — especially when it is not possible for the tests to fully represent accident conditions.

(o) Requirement 32: Outage management. The outage management may not consider simpler designs, fewer SSCs, replacement of major components, ability to perform maintenance activities remotely, remote surveillance, inspection and testing techniques, access to the equipment, outage frequency and, in some cases, outage work at the factory.

4.5.2. IAEA Safety Standards Series No. SSG-28, Commissioning for Nuclear Power Plants

4.5.2.1. Areas of applicability

SSG-28 [47] is generally applicable to EIDs. Some gaps were identified and are described below.

4.5.2.2. Areas of non-applicability

No areas of non-applicability were identified.

4.5.2.3. Identified gaps and areas for additional consideration

The following points may not be addressed in SSG-28 [47]:

(a) Commissioning of modules that may include integrated SSCs.
(b) How to consider the impact of the addition of new modules to the commissioning of shared systems in a multi-module plant, given that shared SSCs may require certain commissioning activities to take place as the first and subsequent modules are installed and put into service. This may depend on how shared SSCs are credited for plant safety and how this is affected by the addition of new modules.
(c) The extent to which commissioning activities on the first module can be credited for future reactor modules in the same or other NPPs.
(d) How commissioning of a module can demonstrate and verify its compatibility with the existing modules.
(e) Further on-site commissioning tests that may need to be performed to check that the results obtained off-site are valid for the plant. In addition, commissioning at the factory may sometimes involve the site operating organization.
(f) The performance of shared systems when adding reactor modules and whether additional or new or repeated commissioning tests may be needed. This may also require consideration if a design is being replicated at multiple sites.

4.5.3. IAEA Safety Standards Series No. SSG-76, Conduct of Operations at Nuclear Power Plants

4.5.3.1. Areas of applicability

SSG-76 [48] is generally applicable to EIDs. Some gaps were identified and are described below.

4.5.3.2. Areas of non-applicability

No specific areas of non-applicability were identified.

4.5.3.3. Identified gaps and areas for additional consideration

The following gaps were identified:

(a) The operation of multiple reactor modules from the same MCR and plant systems shared by all or some reactor modules are not covered by SSG-76 [48].
(b) Use of autonomous and remote monitoring and intervention technologies in plant operations both on-site and off-site are not discussed.
(c) Assurance of human factors and human performance in consideration of novel operator aids/tools and communications equipment between the plant and remote control centre operations facilities are not covered by this guidance.
(d) The Phase 2 report of the SMR Regulators' Forum, Working Group on Manufacturing, Construction, Commissioning and Operation [126], states:

 "A factory-fuelled and sealed transportable reactor module is a special case because it represents a significant nuclear island component manufactured and likely tested, or even commissioned, to some degree off-site…In addition, some vendors are proposing that a vendor/assembler factory may consider low-power nuclear testing of the reactor module before the module leaves the factory…The challenge presented in this case is that these testing activities would require an operating license in the factory's jurisdiction".

 Hence, there may be at least two operators, one at the factory and another at the deployment site. The guidance does not appear to cover these considerations.

(e) The same report [126] also indicates some aspects of fleet operations that the guidance does not seem to consider. Some of these aspects are:

"As a fleet of SMRs develops over time, there will be operational experience (OPEX) from the first of a kind, and subsequently from later Nth of a kind, which should be gathered, collated, and analysed…

"A comprehensive OPEX management system will mean that Licensees for subsequent SMRs in a fleet can be assured that the design for their Nth of a kind will take account of OPEX from existing sites.

…….

"Prompt inclusion of OPEX (from other SMRs in the fleet) into maintenance activities for all the SMRs".

4.5.4. IAEA Safety Standards Series No. SSG-72, The Operating Organization for Nuclear Power Plants

4.5.4.1. Areas of applicability

SSG-72 [49] is generally applicable to EIDs. Some gaps are presented below.

4.5.4.2. Areas of non-applicability

No areas of non-applicability were identified.

4.5.4.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) Considerations of the operation of a fleet of EIDs (by the same operator or different operators) is not addressed in the Safety Guide. SSG-72 [49] considers that all safety related operator actions are carried out by staff located at the site. EID proponents are considering different operational models that may have some staff located at the site for specific NPP activities, and potential 'fleet service facility' operating staff with a role of providing operational functions and support to the site either via traditional approaches, such as training, or via direct intervention through remote I&C connections to each plant.

(b) For some EIDs, the use of 'off the shelf' manufacturing means that the site operator may not be known at the time of manufacturing. Furthermore, for TNPPs the responsible organization during transportation may be different from the operating organization. At present, the roles and responsibilities of the different types of organization are not clear.

4.5.5. IAEA Safety Standards Series No. SSG-70, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants

4.5.5.1. Areas of applicability

SSG-70 [50] is generally applicable to EIDs.

4.5.5.2. Areas of non-applicability

While the OLCs and operating procedures will be different for different reactor types and designs, no areas of non-applicability were identified.

4.5.5.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) This Safety Guide is predicated on existing NPPs and does not adequately address first of a kind NPPs. The derivation of OLCs for any novel technology, in particular for a first of a kind NPP, is particularly challenging in view of the overlapping technological uncertainties and the lack of sufficient operating experience. Guidance does not currently assist the user to understand that a conservative approach may be necessary in establishing OLCs and operating procedures and that these conservative measures can be revisited and adjusted (with justification) when operating experience is accumulated. For a first of a kind NPP, experience in the technical assessment of advanced reactors shows that commissioning activities may be needed to complete the verification of plant characteristics and performance. This may include specialized tests, such as inducing transients that the NPP is designed to cope with. This would need to be addressed in the safety case, and additional safety and control provisions may need to be established.
(b) Along the same lines as the above for the first of a kind NPP, the development of operating procedures will need to include more frequent

use of human performance error reduction tools[29] by operators and consideration of backout provisions[30]. Early revision cycles for operating and maintenance procedures will need to be carefully integrated by staff responsible for human performance training and commissioning to ensure that error prone situations are avoided until sufficient experience with the physical plant is achieved (there may be differences between the real plant and the simulations used in training). There is also a gap regarding the use of remote surveillance and testing, so additional guidance is required to prevent situations where there is a discrepancy between site measured parameters and those measured remotely to ensure that test and surveillance results give an accurate representation of plant values.

4.5.6. IAEA Safety Standards Series No. SSG-71, Modifications to Nuclear Power Plants

4.5.6.1. Areas of applicability

SSG-71 [51] is generally applicable to EIDs.

4.5.6.2. Areas of non-applicability

No areas of non-applicability were identified.

4.5.6.3. Identified gaps and areas for additional consideration

The following gaps were identified:

(a) Some EIDs are being proposed as ‘fleets’ of standardized NPPs. However, history from the nuclear sector, as well as many other high technology industries, has shown that standardization is rarely achieved until several

---

[29] ‘Error reduction tools’ in this context means human performance error reduction tools, such as peer checking. The point being that, as the system is first of a kind, the operations and maintenance of the system will be unfamiliar to personnel. Hence, the level of detail within the operating and maintenance procedures may need to be more comprehensive than for systems with which personnel are more familiar. It follows that there is a need to include more human performance tools within the procedure (such as place keeping and peer checking) to minimize the likelihood of operators making a mistake.

[30] In this context, ‘backout provision’ means that if an unexpected condition occurs when operating the equipment following the procedure, then the procedure contains clear guidance on how to go back to a known state, to reconsider/proceed more carefully through the operating sequence, and to try again.

facilities have been built and sufficient feedback has been accumulated and analysed. The design is normally modified and incorporated in the ‘next build’, where the cycle begins again. When a fleet is being deployed, this means the relevant entity (EID developer, licensee or operator) will need to decide to what extent to reconcile the earlier facilities with the updated design on the basis of lessons learned. This will need to be coordinated across the fleet in a controlled manner to keep tight control of the configuration at each facility. Where centrally coordinated (fleet wide) modifications and maintenance are used, this entity will need access to highly accurate local operational and configuration information and to have strong interfaces with the operator to prevent inadvertently introducing modifications that depart from the site specific safety case.

(b) For a first of a kind facility (and other first few of a kind), the types and natures of modifications, and their justification as necessary or not, may represent a significant challenge.

4.5.7. IAEA Safety Standards Series No. SSG-73, Core Management and Fuel Handling for Nuclear Power Plants

4.5.7.1. Areas of applicability

SSG-73 [52] is generally applicable to EIDs.

4.5.7.2. Areas of non-applicability

No areas of non-applicability were identified.

4.5.7.3. Identified gaps and areas for additional consideration

The following gaps were identified:

(a) Fuel loading, reloading, unloading or refuelling when it is performed at a separate off-site facility;
(b) Handling of specific issues and hazards when irradiated fuel is not stored on-site;
(c) The consequences of novel fuel types (e.g. HTGR fuel) and fuel defects;
(d) Situations where different organizations are responsible for fuel loading and unloading, regardless of whether this takes place on-site or off-site.

4.5.8. IAEA Safety Standards Series No. SSG-74, Maintenance, Testing, Surveillance and Inspection in Nuclear Power Plant

4.5.8.1. Areas of applicability

SSG-74 [53] is generally applicable to EIDs.

4.5.8.2. Areas of non-applicability

No areas of non-applicability were identified.

4.5.8.3. Identified gaps and areas for additional consideration

The following gaps were identified:

(a) Implementation of a programme that involves multiple reactor modules within a single facility and the use of shared and/or common SSCs. This requires additional programmatic elements to prevent human induced events and maintenance impacts on adjacent reactor modules or that potentially impair safety related functions of shared systems.
(b) Use of technologies for autonomous and/or remote monitoring for testing, inspection, surveillance and maintenance.
(c) Maintenance, surveillance and periodic testing of systems supporting inherent and passive safety functions.
(d) Compact designs that cannot accommodate physical inspections or bulky inspection equipment. For example, maintenance, surveillance, inspections and periodic testing of equipment integrated within the reactor vessel. The compact nature of some EIDs may limit the ability to perform the necessary inspections and maintenance.
(e) Ability for on-site measurement collection to manage situations where there has been a remote plant data system failure (e.g. inferring data using factors such as fluence and chemistry, when a direct measurement is not possible).
(f) Availability of staff needed for local site activities in the case of remote activities or autonomous operations failure.
(g) Maintenance, testing, surveillance and inspection of a fleet of standardized technologies. This will need to be coordinated, and the lessons learned from these activities need to be shared. This may be a challenge for an EID fleet where the operators have not yet been identified.

4.5.9. IAEA Safety Standards Series No. SSG-75, Recruitment, Qualification and Training of Personnel for Nuclear Power Plants

4.5.9.1. Areas of applicability

SSG-75 [55] is generally applicable to EIDs.

4.5.9.2. Areas of non-applicability

No areas of non-applicability were identified.

4.5.9.3. Identified gaps and areas for additional consideration

The current guidance does not include consideration of additional training strategies to address specific EID features, including the following:

(a) The operation of multiple reactor modules from the same MCR is a departure from current practice on NPPs.
(b) Some EID fleets may share technical staff, so an individual EID may have limited numbers of staff or even no staff present during normal operation. This is a departure from current practice.
(c) EIDs that are located in remote regions may need to operate with an unreliable or limited grid connection, so a black start capability becomes essential; an ability to tolerate long periods of extreme weather conditions may also be needed.
(d) The sharing of staff between several EIDs may increase the need for remote monitoring of the plant and for the execution of surveillance, testing, inspection and maintenance activities.
(e) The likelihood that some EIDs will be constructed, fuelled and defuelled under factory conditions raises issues regarding the handling and storage of both fresh and irradiated fuel at such facilities and on how to manage the throughput of a series of reactors that, while similar, may not be identical.

4.5.10. IAEA Safety Standards Series No. SSG-13, Chemistry Programme for Water Cooled Nuclear Power Plants

4.5.10.1. Areas of applicability

SSG-13 [56] is generally applicable to water cooled EIDs.

4.5.10.2. Areas of non-applicability

The Safety Guide is not applicable to non-water cooled EIDs.

4.5.10.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) Paragraphs 4.14–4.41 include specific guidance on primary coolant chemistry in different WCR types. The guidance was derived for fuel designs as of 2011 and might not be fully applicable or inclusive of novel fuel designs considered for WCRs, including SMRs, such as certain ATF concepts.
(b) Paragraphs 4.42–4.49 include guidance on secondary circuit water chemistry, which is also applicable to, but might not be comprehensive enough for, non-water cooled EIDs if they utilize water in a secondary or tertiary circuit (such as SFRs) to generate subcritical steam. The guidance is not applicable to supercritical water circuits.

4.5.11. IAEA Safety Standards Series No. SSG-48, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants

4.5.11.1. Areas of applicability

SSG-48 [57] is generally applicable to EIDs.

4.5.11.2. Areas of non-applicability

No areas of non-applicability were identified.

4.5.11.3. Identified gaps and areas for additional consideration

A gap was identified regarding the chemistry of all fluid filled systems within a chemistry programme, including the different primary, secondary and tertiary circuit coolants and component materials.

4.5.12. IAEA Safety Standards Series No. SSG-50, Operating Experience Feedback for Nuclear Installations

4.5.12.1. Areas of applicability

SSG-50 [58] is generally applicable to EIDs.

4.5.12.2. Areas of non-applicability

No areas of non-applicability were identified.

4.5.12.3. Identified gaps and areas for additional consideration

No gaps were identified. However, in the context of EIDs, the term ‘operating experience’ may need to be expanded to include feedback from design and safety assessment activities.

4.5.13. IAEA Safety Standards Series No. SSG-77, Protection Against Internal and External Hazards in the Operation of Nuclear Power Plants

4.5.13.1. Areas of applicability

SSG-77 [59] is generally applicable to EIDs.

4.5.13.2. Areas of non-applicability

No areas of non-applicability were identified.

4.5.13.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) The impact of the use of technologies for autonomous operation, remote monitoring and testing, or both are not covered in SSG-77 [59]. Examples include the following:
   (i) Provisions for protection from site specific hazards for cases where autonomous operation with or without local on-site staff, coupled with remote control and monitoring, are not covered;
   (ii) Consideration of how hazards protection can still be locally assured before the arrival of off-site resources.

(b) Appendix I of SSG-77 [59] does not consider the type of reactor coolant, the type of combustible material that might be used in an EID, the common fire detection and firefighting equipment, or the coordination of firefighting activities that would be required for a multi-module NPP. Similar issues can be identified for other hazards.

4.5.14. IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants

4.5.14.1. Areas of applicability

Recommendations in SSG-54 [60] on general aspects of accident management programmes, their scope and objectives, as well as their development, implementation and execution, are largely written in a technology neutral manner and are therefore mostly applicable to EIDs.

4.5.14.2. Areas of non-applicability

No specific areas of non-applicability were identified in the approach to development, implementation and execution. However, a few recommendations address very specific aspects of severe accident phenomenology and progression for WCRs, such as zircaloy oxidation, hydrogen explosion, steam explosion and high pressure melt ejection. Similarly, several recommendations refer to fuel rods, core damage, containment, reactor pressure vessel, containment venting system, water for DHR and core exit temperature.

4.5.14.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) Since the definitions of DECs and severe accident are largely based on WCRs, the terms applicable to EID technologies regarding the failure of defined fission product barriers are missing. Definitions of preventive and mitigatory domains for EIDs are also not included in in SSG-54 [60].
(b) Guidance for the use of a graded approach in accident management considerations is not provided.
(c) The operating organization and the emergency response organizations and their practical arrangements for remotely controlled EIDs with non-permanent staff on-site are not covered by the guide. Other items not covered are aspects of the accessibility of the site in the case of extreme external hazards, the role of accident progression, the usage of autonomous

technologies in decision making, local and regional competence and skills, and the use of relevant terminology (e.g. main control room, technical support centre, operating personnel for mitigatory actions). In addition, the current understanding, based on traditional WCRs, of on-site/off-site considerations (e.g. decisions, actions) is expected to be rather different for remotely controlled reactors, but this is not considered.

(d) The objectives of an accident management programme are not elaborated for EID technologies. Considerations related to paras 2.14 (accident programme objectives), 3.53 (aspects of containment venting), 3.88 (capability of equipment and margins to failure), 3.100 and 3.101 (analysis of plant capabilities for in-vessel and ex-vessel phases of a severe accident, respectively) of SSG-54 are specific to WCRs and fuel rods. These considerations may still be applicable to some EIDs, but their equivalents for other EIDs are not covered.

(e) Several recommendations provide specific examples for current WCR designs. Technology specific examples for various EIDs are missing in SSG-54, where guidance is informed by experience with WCR technology.

(f) There is a lack of guidance on the preventive domain and emergency operating procedures compared with the mitigatory domain and severe accident management guidelines. Technologically neutral guidance for EIDs is not covered.

4.6. NUCLEAR FUEL CYCLE FACILITIES

4.6.1. IAEA Safety Standards Series No. SSR-4, Safety of Nuclear Fuel Cycle Facilities

4.6.1.1. Areas of applicability

Most of the safety requirements established in SSR-4 [20] are technology neutral and generally applicable to all types of nuclear fuel cycle facility, including those associated with EIDs. Some of the requirements established in SSR-4 are specific to a certain type of nuclear fuel cycle facility, such as nuclear fuel reprocessing plants or MOX fuel fabrication facilities.

4.6.1.2. Areas of non-applicability

No areas of non-applicability were identified.

4.6.1.3. Identified gaps and areas for additional consideration

Depending on the particular fuel type and design, further technology specific requirements may need to be established in SSR-4. These would be associated for example with the following:

(a) Enrichment levels beyond 6% ^{235}U;
(b) Specific chemical or fire hazards;
(c) Additional requirements during commissioning of fuel cycle facilities for EIDs on an industrial scale, noting the lack of previous experience in this respect;
(d) Training and qualification of personnel where new technologies are implemented;
(e) Design of specific systems that are not currently present in existing nuclear fuel cycle facilities.

4.6.2. IAEA Safety Standards Series No. SSG-27 (Rev. 1), Criticality Safety in the Handling of Fissile Material

4.6.2.1. Areas of applicability

Recommendations in SSG-27 (Rev. 1) [53] related to criticality safety in the handling of fissile material are general and not limited to any specific technology. Therefore, this Safety Guide is fully applicable to EIDs.

4.6.2.2. Areas of non-applicability

No areas of non-applicability were identified.

4.6.2.3. Identified gaps and areas for additional consideration

No gaps were identified.

4.6.3. IAEA Safety Standards Series Nos SSG-5 (Rev. 1), Safety of Conversion Facilities and Uranium Enrichment Facilities, SSG-6, Safety of Uranium Fuel Fabrication Facilities, SSG-7 (Rev. 1), Safety of Uranium and Plutonium Mixed Oxide Fuel Fabrication Facilities, and SSG-42, Safety of Nuclear Fuel Reprocessing Facilities

4.6.3.1. Areas of applicability

Specific Safety Guides SSG-5 (Rev. 1) [61], SSG-6 [62], SSG-7 (Rev. 1) [63] and SSG-42 [64] provide recommendations for certain types of nuclear fuel cycle facility. The area of applicability is therefore limited to the named facility type, as follows:

(a) SSG-5 (Rev. 1) applies to the handling, processing and storage of depleted, natural and low enriched uranium (LEU) with a ^{235}U concentration of no more than 6%, which could be derived from natural, high enriched, depleted or reprocessed uranium. Conversion facilities covered by SSG-5 apply to the conversion of uranium concentrate to UF_6. A gas centrifuge process is considered for uranium enrichment facilities.
(b) SSG-6 deals specifically with the handling, processing and storage of LEU that has a ^{235}U concentration of no more than 6%, derived from natural, high enriched or reprocessed uranium; it does not cover facilities that handle uranium metal fuels. Completed fuel assemblies (e.g. fuel assemblies for pressurized water reactors, boiling water reactors, heavy water reactors (such as CANDU reactors), advanced gas cooled reactors) are stored at the fuel fabrication facility before being transported to the NPP. Such a storage facility is considered to be part of the fuel fabrication facility.
(c) SSG-7 (Rev. 1) applies to the handling, processing and storage of: (i) plutonium oxide; (ii) depleted, natural or reprocessed uranium oxide; and (iii) MOX manufactured from plutonium oxide and uranium oxide for use as a feed material to form MOX fuel rods and assemblies in WCRs and fast breeder reactors. The fuel fabrication processes covered by SSG-7 (Rev. 1) are dry processes.
(d) SSG-42 provides recommendations that apply to reprocessing plants using the PUREX process to reprocess fuels containing uranium and plutonium on a commercial scale. This Safety Guide does not specifically address THOREX. However, the similarity between aqueous reprocessing methods means that these recommendations will apply, with suitable adjustments, to plants reprocessing many types of nuclear fuel. The fuel reprocessing processes covered by SSG-42 are a mixture of high and low hazard chemical

and mechanical processes, including high hazard fine particulate processes, and processing involving hazardous solid, liquid, gaseous and particulate (dry, air and water-borne) wastes and effluents.

4.6.3.2. Areas of non-applicability

The following areas of non-applicability were identified:

(a) SSG-5 (Rev. 1) is not applicable to any conversion or uranium enrichment processes that are not listed in Section 4.6.3.1. Moreover, for LEU with enrichment higher than 6%, SSG-5 (Rev. 1) currently does not apply; however, extension would be possible through a revision by amendment.
(b) SSG-6 is not applicable to fuel manufacturing facilities that: (i) process LEU with enrichment higher than 6%; (ii) use fissile materials other than uranium; or (iii) manufacture uranium metal fuels.
(c) SSG-7 (Rev. 1) does not apply to any other processes except the dry process for MOX fuel manufacturing.
(d) SSG-42 provides recommendations for fuel reprocessing plants that are not applicable to processes other than PUREX or to reprocessing fissile material other than uranium and plutonium.

4.6.3.3. Identified gaps and areas for additional consideration

Safety Guides SSG-5 (Rev. 1), SSG-6, SSG-7 (Rev. 1) and SSG-42 address mature and industrialized fuel fabrication processes and may not cover some fuel types and designs. New Safety Guides for the manufacturing of advanced fuels may need to be developed in future, but perhaps not before the various technologies are mature and industrialized. Until this point is reached, the guidance in the above mentioned Safety Guides and in SSG-43 may be used, as applicable.

4.6.4. IAEA Safety Standards Series No. SSG-43, Safety of Nuclear Fuel Cycle Research and Development Facilities

4.6.4.1. Areas of applicability

Recommendations provided in SSG-43 [65] apply to any R&D facilities that are associated with the nuclear fuel cycle. This Safety Guide focuses specifically on the safe design, construction, commissioning, operation and preparation for decommissioning of R&D facilities. Its scope is limited to the safety of the R&D facility, the protection of workers and the public, and the management of any wastes generated. SSG-43 could be applied to demonstration or prototype facilities built for fuel cycle facilities of EIDs. It contains radiological as well as chemical hazards and is not technology specific. This Safety Guide could be especially applicable in the early phase of fuel cycle facilities of EIDs, before standardized fuel types and associated facilities are developed on an industrial scale.

4.6.4.2. Areas of non-applicability

No areas of non-applicability were identified. SSG-43 does not apply to industrial facilities and pilot plants.

4.6.4.3. Identified gaps and areas for additional consideration

Specific hazards arising from EIDs may not be covered by SSG-43 [65] and would be best identified once the precise type and design of fuel and its manufacturing/reprocessing processes are known.

4.7. RADIATION PROTECTION AND SAFETY

4.7.1. IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards

4.7.1.1. Areas of applicability

The requirements in GSR Part 3 [10] are built on the three general principles of radiation protection — justification, optimization and dose limitation — and application of the graded approach. Although EIDs are not specifically mentioned, many of the requirements in GSR Part 3 clearly apply. Of most relevance are those addressing the protection of workers, the public and the environment in planned exposure situations and in emergency exposure situations. The four overarching requirements on emergency exposure situations presented in paras 4.2–4.21 of GSR Part 3 are covered in greater detail in GSR Part 7 and discussed in Section 4.12.1.

4.7.1.2. Areas of non-applicability

Requirements related to medical exposures, consumer products and exposure due to radionuclides in commodities are not applicable to reactors, including EIDs.

4.7.1.3. Identified gaps and areas for additional consideration

No gaps were identified.

4.7.2. IAEA Safety Standards Series No. GSG-7, Occupational Radiation Protection

4.7.2.1. Areas of applicability

GSG-7 [66] provides guidance on fulfilling the requirements in GSR Part 3 [10] for the protection of workers in planned, emergency and existing exposure situations. It provides general guidance on the development of occupational radiation protection programmes, as appropriate, for the sources of radiation likely to be encountered in the workplace, to assist the management in fulfilling its responsibility for the protection and safety of workers. Detailed guidance is also provided on the monitoring and assessment of workers' exposure due to external radiation sources and from intakes of radionuclides.

4.7.2.2. Areas of non-applicability

No areas of non-applicability were identified.

4.7.2.3. Identified gaps and areas for additional consideration

No gaps have been identified. Potential areas for additional consideration relate to occupational radiation protection issues that could arise from the areas of novelty in design, operation, maintenance and decommissioning. These areas can impact activities carried out by the staff and, hence, affect the associated radiation doses to workers.

4.7.3. IAEA Safety Standards Series No. GSG-8, Radiation Protection of the Public and the Environment

4.7.3.1. Areas of applicability

GSG-8 [67] provides generic guidance and recommendations related to planned exposure situations, existing exposure situations and emergency exposure situations. Some guidance for existing exposure situations is applicable to EIDs.

4.7.3.2. Areas of non-applicability

No areas of non-applicability were identified.

4.7.3.3. Identified gaps and areas for additional consideration

No gaps were identified.

4.7.4. IAEA Safety Standards Series No. GSG-9, Regulatory Control of Radioactive Discharges to the Environment

4.7.4.1. Areas of applicability

GSG-9 [68] provides generic recommendations and guidance for discharges from normal operations of facilities and activities and is applicable to EIDs.

4.7.4.2. Areas of non-applicability

No areas of non-applicability were identified.

4.7.4.3. Identified gaps and areas for additional consideration

Guidance on regulating discharges to the environment from EIDs that are installed underground (which is expected for some designs) is not included.

4.7.5. IAEA Safety Standards Series No. GSG-10, Prospective Radiological Environmental Impact Assessment for Facilities and Activities

4.7.5.1. Areas of applicability

GSG-10 [69] provides generic recommendations and guidance and is applicable to EIDs.

4.7.5.2. Areas of non-applicability

No areas of non-applicability were identified.

4.7.5.3. Identified gaps and areas for additional consideration

Aspects of the radiological environmental impact assessment for EIDs installed underground (which is expected for some designs) are not included.

4.7.6. IAEA Safety Standards Series No. RS-G-1.7, Application of the Concepts of Exclusion, Exemption and Clearance

RS-G-1.7 is currently under revision. The revised Specific Safety Guide will focus on the application of the concept of clearance [70].

4.7.6.1. Areas of applicability

This Safety Guide provides generic recommendations and guidance and is applicable to EIDs.

4.7.6.2. Areas of non-applicability

The concepts of exclusion and exemption are not relevant to reactors and, therefore, are not relevant to EIDs.

4.7.6.3. Identified gaps and areas for additional consideration

No gaps were identified.

4.8. MANAGEMENT OF RADIOACTIVE WASTE AND SPENT FUEL

A general observation from the review is that, so far, developers of EIDs have provided relatively little detailed information on the management of waste and spent fuel from EIDs (e.g. NR-T-1.18 [128]). The review findings summarized below highlight the identified gaps and areas for additional consideration.

4.8.1. IAEA Safety Standards Series No. GSG-1, Classification of Radioactive Waste

4.8.1.1. Areas of applicability

GSG-1 [71] is sufficiently general to apply to the waste that could be generated from EIDs.

4.8.1.2. Areas of non-applicability

No areas of non-applicability were identified.

4.8.1.3. Identified gaps and areas for additional consideration

The following gaps were identified:

(a) A future update of the standard might address some novel waste types and explain how they could fit into the classification.
(b) The small size of SMRs suggests that operators may consider it beneficial to dispose of an entire reactor module, including the fuel. An application to dispose of SNF in this way could be considered on its own merits (according to the risks) without a strict reference to the IAEA waste classification. It may be helpful for GSG-1 to consider such a situation.

4.8.2. IAEA Safety Standards Series Nos GSR Part 5, SSG-40, SSG-41, SSG-45, WS-G-6.1 and GSG-3 on predisposal management of radioactive waste

4.8.2.1. Areas of applicability

The IAEA Safety Standards Series publications GRS Part 5 [12], SSG-40 [73], SSG-41 [74], SSG-45 [75], WS-G-6.1 [76] and GSG-3 [77] on waste management are sufficiently general to apply to all waste from EIDs.

However, when examining them in detail, there are a few areas that may benefit from additional consideration, as indicated in Section 4.8.2.3.

4.8.2.2. Areas of non-applicability

The review did not identify any areas of non-applicability to wastes from EIDs.

4.8.2.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) GSR Part 5 [12] mentions the need for radioactive waste to be stored in a “passive, safe condition”, but passive safety is not itself a requirement. This contrasts with the publication addressing waste disposal (SSR-5 [21]), where passive safety is a requirement. The generation and long term storage of radioactive waste at multiple sites could, if permitted, have implications for safety, on account of the possibly lower level of supervision at the site — for example, of a single EID or SMR — than what would be achievable at the site of a large NPP. For this reason, it might be worth considering whether passive safety in predisposal management (including storage) should be upgraded to a requirement. While this point is not restricted to EIDs, its significance could be raised if SMRs were to be spread widely across a territory.

(b) The development and deployment of EIDs could give rise to many different types of spent fuel. A possible consequence of this is an increasing dependence on direct disposal because of a lack of availability of appropriate reprocessing facilities. It is likely that metal fuel will not be suitable for direct disposal; in this case, some form of processing (as opposed to reprocessing) could be necessary to allow its disposal. SSG-40 [73] provides guidance on the predisposal management of radioactive waste packages and SNF declared as waste. When SSG-40 mentions ‘processing’, however, this is always in the context of non-fuel waste — that is, there appears to be an unspoken assumption that ‘processing’ (as opposed to reprocessing) does not apply to SNF. It may be helpful, therefore, to revise SSG-40 to recognize the possibility of processing (not reprocessing) of SNF for disposal.

(c) To avoid having radioactive waste stored at many sites, operators may wish, or be required by the regulatory body, to use a centralized facility. Existing IAEA guidance (SSG-45, WS-G-6.1 [75, 76]) expresses a preference for centralized storage of waste, but the guidance comprises just two paragraphs (para. 5.3 of WS-G-6.1 [76] and para. 4.80 of SSG-45 [75]). This guidance could sensibly be elaborated on and could elucidate the advantages and disadvantages of such a strategy and related matters, such as site selection for a centralized facility for the storage of waste.

(d) The operational safety of waste and SNF management facilities at EID sites could be impacted by the use of EIDs for cogeneration on the same site. This is not always well covered by the existing Safety Standards. For example, Requirement 8 of GSR Part 4 (Rev. 1) [11] (assessment of site characteristics) calls for human induced external hazards, such as those arising from industrial activities, to be identified, but mention of external industrial hazards in GSG-3 [77] is limited to an annex, which is not an official part of the guidance.

(e) While sequential construction of power reactors at a single site is not new, GSG-3 [77] does not address this point specifically.

(f) Although all the waste generated from EIDs may be managed in accordance with the IAEA safety standards, it might be beneficial to operators and regulatory bodies to collect further information on the possibilities for waste minimization and on the novel waste types that might be generated from EIDs (see Section 3). This could be done with a view to the possible future development of guidance on the safe management of such waste. This might be addressed by updating SSG-41 [74] to cover the new waste types. In the meantime, the development of a technical document describing the wastes and options for their management would be a logical step.

4.8.3. IAEA Safety Standards Series No. SSG-15 (Rev. 1), Storage of Spent Nuclear Fuel

4.8.3.1. Areas of applicability

Safety Guide SSG-15 (Rev. 1) [78] is sufficiently general to be applied to all SNF that could be generated from EIDs. However, there are a few areas that may benefit from additional consideration, as indicated in Section 4.8.3.3.

4.8.3.2. Areas of non-applicability

SSG-15 (Rev. 1) [78] assumes that fuel will be presented in the form of fuel assemblies. This is entirely appropriate in the context of current reactors but would not be applicable to HTGR pebbles or MSR fuel. Further, in the case of fuel or other components removed from an SFR, the measures described by SSG-15 (Rev. 1) would not be adequate, because of the need to prevent sodium, which may be adhering to the fuel or other components, from reacting with the air.

4.8.3.3. *Identified gaps and areas for additional consideration*

The following gaps and areas for additional consideration were identified:

(a) To avoid having SNF stored at many sites, operators may wish, or be required by the regulatory body, to use a centralized storage facility. SSG-15 (Rev. 1) [78] mainly discusses the storage of SNF at the site of its generation. An area that is not covered is the possibility of centralized storage, including the advantages and disadvantages of such a strategy and related matters, such as site selection for a centralized storage facility for SNF.

(b) A feature of many EIDs is the use of higher enrichment (up to 20%) HALEU fuel to facilitate longer fuel life than traditional fuels. This will increase the risk of unwanted criticality. Criticality safety during storage of SNF is addressed in SSG-15 (Rev. 1) [78] and, in general, is well covered. However, revisions may be considered (e.g. in appendix I) to include two aspects of HTGR fuel, namely the presence of a potent moderator and the particulate (and therefore shape changing) nature of pebble bed fuel; further, SSG-15 (Rev. 1) could refer to SSG-27 (Rev. 1) [53].

4.8.4. IAEA Safety Standards Series Nos SSR-5, SSG-1, SSG-14, SSG-23, SSG-29 and SSG-31 on the disposal of radioactive waste

4.8.4.1. *Areas of applicability*

The IAEA Safety Standards Series publications on radioactive waste management, SSR-5 [21], SSG-1 [79], SSG-14 [81], SSG-23 [82], SSG-29 [83] and SSG-31 [80], are sufficiently general to apply for all waste that could be generated from EIDs. For some of these Safety Standards, there are no implications for EIDs. For example, SSG-1 [81], is concerned with the safe disposal of disused sealed radioactive sources (i.e. not waste that could be generated from EIDs). SSG-23 [82] provides recommendations on the safety case and safety assessment for radioactive waste disposal, and these would be applicable to the disposal of waste from EIDs in a disposal facility. SSG-31 [80] provides recommendations on the monitoring and surveillance of radioactive waste disposal facilities, and these would be applicable to the disposal of waste from EIDs in a disposal facility. However, when examining some of the other safety standards in detail, there are a few areas that may benefit from additional consideration, as indicated in Section 4.8.4.3.

4.8.4.2. Areas of non-applicability

The review did not identify any areas of the waste related standards that would not be applicable to wastes from EIDs.

4.8.4.3. Identified gaps and areas for additional consideration

The following gaps were identified:

(a) SSR-5 [21] deals with the disposal of radioactive waste. Although the requirements established in SSR-5 apply to all the waste that could be generated from EIDs, the disposal of entire reactors was not considered as a possibility during the development of SSR-5. Therefore, there could be benefit in considering whether and how this might be achieved while respecting the safety requirements for disposal, such as those for multiple barriers and safety functions, for containment and for waste acceptance criteria (WAC).
(b) In SSG-29 [83], the discussion in paras 6.29–6.38 rightly emphasizes the role of WAC in assuring safety and the importance of prior regulatory oversight. It may, however, be helpful to add that there may be circumstances where the prevailing WAC are not applicable and, instead, an exceptional argument could be made to, and special permission sought from, the regulatory body for disposal of a specific item of waste, such as a disused reactor vessel.
(c) As noted in relation to GSG-1 [71], the small size of SMRs suggests that operators may consider it beneficial to dispose of an entire reactor module, including the fuel. It may be helpful if the guidance on WAC (e.g. paras 6.36–6.41 of SSG-14 [81]) was revised to address such possibilities and the related questions. For example, clarification on how an operating organization at a disposal facility could establish WAC for entire reactors, and how it could assess compliance with the WAC, may be needed.

4.8.5. IAEA Safety Standards Series No. GSG-16, Leadership, Management and Culture for Safety in Radioactive Waste Management

4.8.5.1. Areas of applicability

GSG-16 [72] is applicable to EIDs.

4.8.5.2. Areas of non-applicability

No areas of non-applicability were identified.

4.8.5.3. Identified gaps and areas for additional consideration

No gaps were identified.

4.9. DECOMMISSIONING

This section deals with four safety standards relevant to decommissioning. One decommissioning related Safety Guide was screened out, namely SSG-49, Decommissioning of Medical, Industrial and Research Facilities [129], which excludes nuclear fuel cycle facilities from its scope.

4.9.1. IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities

4.9.1.1. Areas of applicability

GSR Part 6 [13] is sufficiently general to apply directly to EIDs.

4.9.1.2. Areas of non-applicability

In the case of a TNPP or SMR that is sent to a central fuel removal facility or dismantling facility, the reactor operator’s decommissioning activities would be greatly reduced and, perhaps, limited to decontamination and removal of low level wastes. In contrast, the operator of the centralized dismantling facility will have a much wider range of activities to perform but, because these will form part of the normal ongoing, and possibly long term, operation of the centralized facility, they cannot be classified as decommissioning tasks. Instead, these are predisposal tasks and, as such, are covered by GSR Part 5 [12] rather than GSR Part 6 [13].

4.9.1.3. *Identified gaps and areas for additional consideration*

The following gaps were identified:

(a) The definition of facilities used in GSR Part 6 [13] (see Section 4.9.1.1) would appear to exclude a marine based TNPP. Hence, the scope of GSR Part 6 may not include this type of NPP.

(b) The definition of immediate dismantling used in the standard (para. 1.9 of GSR Part 6) states that: "structures, systems and components of a facility containing radioactive material are removed and/or decontaminated". Use of the word 'removed' here seems to allow the possibility that some components or even a whole reactor could be taken intact from the site and transferred to a specialized dismantling and waste management facility. Such a transfer could require responsibility for the material to be changed to a new licensee, however, and here the wording of GSR Part 6 is less flexible because the standard seems to assume that any such transfer would occur only if responsibility for the site were taken over by a new entity. For example, para. 4.7 of GSR Part 6 states that: "If the licensee changes during the lifetime of the facility, procedures shall be put in place to ensure the proper transfer of responsibilities for decommissioning to the new licensee." Therefore, there is no guidance for considering if activities that otherwise might be performed on the site could be legitimately conducted elsewhere, even overseas. Depending on the location, regulation of these activities might fall to another regulator; however, as explained above, such activities would probably fall into the scope of GSR Part 5 [12] rather than that of GSR Part 6 [13].

(c) Requirement 8 (on selecting a decommissioning strategy) introduces the possibility that a site may contain more than one (nuclear) facility and calls for interdependencies to be considered. However, the possibility that an NPP could, for example, supply process heat to nearby industrial facilities is not covered.

(d) Requirement 9 of GSR Part 6 places an obligation on the government to establish a mechanism for the funding of decommissioning. The use of TNPPs as a means of providing energy to remote regions will probably result in the development of centralized facilities for reactor dismantling and predisposal waste management that may not be located at the site of deployment. In such cases, decommissioning costs would include payment to the operator of the centralized dismantling facility and, if the central facility was in another State, may need to include currency risk. This may not be covered in the standard.

4.9.2. IAEA Safety Standards Series No. SSG-47, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities

4.9.2.1. Areas of applicability

SSG-47 [84] covers the complete range of nuclear fuel cycle facilities and is non-specific with respect to the type of facility that might be decommissioned or its location; this allows it to be applicable to EIDs. Regarding individual aspects that might be applicable to EIDs:

(a) In para. 7.33, the guide introduces the possibility that some large components might be removed in one piece for storage and processing elsewhere.
(b) It mentions criticality safety for sites where nuclear fuel is present and refers to the applicable Safety Guide. This may become more important if the use of HALEU fuel expands.
(c) Section 6 of SSG-47 discusses the financing of decommissioning and applies equally to current NPPs and EIDs (however, see also Section 4.9.2.3).

4.9.2.2. Areas of non-applicability

No areas of non-applicability were identified.

4.9.2.3. Identified gaps and areas for additional consideration

The following gap was identified: SSG-47 [84] does not consider that when a reactor is transferred for dismantling overseas, future payments will need to consider currency risk.

No other gaps were identified, but some of the examples given in SSG-47 could be usefully expanded as follows:

(a) Paragraph 5.21 of SSG-47 briefly mentions nuclear facilities that might be located close to medical or academic premises. This has some resonance for EIDs located alongside cogeneration facilities, but more direct guidance might be helpful. In para. 5.21, for instance, it is noted that some research reactors could present challenges to decommissioning when being located in a medical facility or on a university campus. The use of EIDs for cogeneration suggests that it would be useful to also include an industrial parallel, such as decommissioning on a site that serves a nearby non-nuclear industrial plant.

(b) To allow for the possibility that TNPPs could be shipped abroad for defuelling and dismantling, in para. 7.33 it could be useful to add that a dismantling facility could be located in another country.
(c) The possibility that decommissioning costs might include payment for dismantling off-site is not mentioned in Section 6 of SSG-47 (financing of decommissioning).

4.9.3. IAEA Safety Standards Series No. WS-G-5.1, Release of Sites from Regulatory Control on Termination of Practices

4.9.3.1. Areas of applicability

This Safety Guide provides guidance on the release of sites (or parts of sites) from regulatory control after a practice has been terminated and on the cleanup of contamination, which is usually a precondition for such release; cleanup is viewed as a decommissioning activity.

WS-G-5.1 [85] applies to every type of facility under decommissioning, with the exception of mine tailings, radioactive waste disposal facilities and intervention situations. It follows that EIDs fall within its scope.

Although the Safety Guide contains references to buildings, structures and environmental media, the text is non-specific about the nature of the facility that is being cleaned up, everything being framed in terms of 'cleanup of the site', and there is no mention of the type of contamination or the method of cleanup. As a result, the guide is applicable to a wide range of facilities, including EIDs.

As with current practice, cleanup of an EID site (or part of it) will invariably occur late in the decommissioning process, after the most heavily contaminated items, such as the reactor, have been removed from the site. This suggests that the introduction of EIDs will have no impact on the guidance provided in WS-G-5.1 [85].

4.9.3.2. Areas of non-applicability

No areas of non-applicability were identified.

4.9.3.3. Identified gaps and areas for additional consideration

No gaps were identified.

4.9.4. IAEA Safety Standards Series No. WS-G-5.2, Safety Assessment for the Decommissioning of Facilities Using Radioactive Material

4.9.4.1. Areas of applicability

WS-G-5.2 [86] applies to all types of facility with the same exceptions as WS-G-5.1 (see Section 4.9.3.1) and is silent on the original purpose or the nature of the facility being decommissioned. It is clear, therefore, that its scope includes EIDs.

4.9.4.2. Areas of non-applicability

No areas of non-applicability were identified.

4.9.4.3. Identified gaps and areas for additional consideration

No gaps were identified.

4.10. LEADERSHIP AND MANAGEMENT FOR SAFETY

4.10.1. IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety

4.10.1.1. Areas of applicability

GSR Part 2 [9] is generally applicable to EIDs.

4.10.1.2. Areas of non-applicability

No areas of non-applicability were identified.

4.10.1.3. Identified gaps and areas for additional consideration

The following potential gaps were identified in GSR Part 2:

(a) Requirement 11 (management of supply chain) does not cover the following areas:
   (i) Oversight and quality control of suppliers in cases where the future owner and operator are not known;

(ii) Guidance for targeted inspection of quality compliance, together with checks of quality assurance arrangements and the management of deviations.

(b) In view of some novel ideas of ownership that have been proposed (e.g. leasing of reactors), the standard may not cover who has the responsibility for safety, especially when the ownership arrangements are made between two Member States.

4.10.2. IAEA Safety Standards Series No. GS-G-3.1, Application of the Management System for Facilities and Activities

4.10.2.1. Areas of applicability

GS-G-3.1 [87] is applicable to EIDs.

4.10.2.2. Areas of non-applicability

No areas of non-applicability were identified.

4.10.2.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) Licensees must exercise oversight and quality control practices over supply chains that are increasingly complex and international and may also include non-nuclear suppliers. Although this aspect is relevant to all nuclear installations, it may need to be reflected in the standard.
(b) Ownership and operating models for fleets of geographically dispersed EIDs (e.g. microreactors, including TNPPs) may diverge from the traditional norm of a single owner/operator/licensee, and this may not be considered by the standard.
(c) The Safety Guide does not cover responsibilities for demonstrating regulatory compliance, suitability, quality control and other relevant characteristics when procuring first of a kind components and services and commercial grade components in cases where the future licensee is not known.

4.10.3. IAEA Safety Standards Series No. GS-G-3.5, The Management System for Nuclear Installations

4.10.3.1. Areas of applicability

GS-G-3.5 [88] is applicable to EIDs.

4.10.3.2. Areas of non-applicability

No areas of non-applicability were identified.

4.10.3.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) Licensee resources for oversight and quality control of suppliers considering different, and possibly more complex, supply chains compared with those traditionally used for WCRs.
(b) Ownership and operating models for fleets of geographically dispersed EIDs (e.g. microreactors, including TNPPs).

4.11. SAFETY ASSESSMENT

4.11.1. IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities

4.11.1.1. Areas of applicability

A high level review of the applicability of GSR Part 4 (Rev. 1) [11] to EIDs was performed with specific focus on the requirements related to safety analysis, as follows:

(a) Requirement 1: Graded approach to safety assessment;
(b) Requirement 11: Assessment of human factors;
(c) Requirement 14: Scope of the safety analysis;
(d) Requirement 15: Deterministic and probabilistic approaches;
(e) Requirement 16: Criteria for judging safety;
(f) Requirement 17: Uncertainty and sensitivity analysis;
(g) Requirement 18: Use of computer codes;
(h) Requirement 19: Use of operating experience data;

(i) Requirement 20: Documentation of the safety assessment;
(j) Requirement 21: Independent verification;
(k) Requirement 24: Maintenance of the safety assessment.

GSR Part 4 (Rev. 1) was developed for all types of nuclear facility and activity in a general manner and, therefore, it is expected to be widely applicable. The review confirmed that for the most part, the requirements are applicable to EIDs.

4.11.1.2. Areas of non-applicability

No areas of non-applicability were identified.

4.11.1.3. Identified gaps and areas for additional consideration

No gaps were identified. Nevertheless, the review concluded that additional guidance would be helpful for the following:

(a) Requirement 1: Graded approach to safety assessment. Additional guidance is needed to elaborate on how the graded approach could simplify and facilitate safety assessment of EIDs.
(b) Requirement 11: Assessment of human factors. Further guidance is needed to explain how to apply human factor rules for multi-module NPPs.
(c) GSR Part 4 (Rev. 1) does not explicitly address first of a kind reactors.

4.11.2. IAEA Safety Standards Series No. SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants

4.11.2.1. Areas of applicability

The general considerations provided in SSG-2 (Rev. 1) [89] are written in a technology neutral way and are applicable to all types of EID. This conclusion refers to the description of the objectives, acceptance criteria, uncertainty analysis, approaches to ensure margins and determination of the source term in deterministic safety analysis (DSA).

In addition, the recommendation to deterministically postulate severe accident conditions (that are relevant to the specific EIDs) regardless of the provisions implemented in the design is applicable. For some EIDs, the possibility of the conditions arising may be considered to have been ‘practically eliminated’ (see para. 3.49 in SSG-2 (Rev. 1)).

With respect to the specification of acceptance criteria, all the recommendations are applicable to water cooled SMRs and the general recommendations on technical and radiological acceptance criteria (paras 4.1–4.5 and 4.16–4.18) and recommendations on radiological acceptance criteria (paras 4.6–4.11) are applicable to EIDs.

Concerning the recommendations in section 5 of SSG-2 (Rev. 1) [89] on the use of computer codes for DSA, it was concluded that they are written in a technology neutral manner and, therefore, are applicable to EIDs. Because of the limited availability of operating experience and detailed design information, as well as the lack of adequately complete experimental data and validation matrices for EIDs, the full scope validation of the codes may be difficult for practitioners until sufficient information is accrued.

The recommendations in section 6 of SSG-2 (Rev. 1) on the need to ensure safety margins and on the conservative and combined approach to the analysis of AOOs and DBAs are applicable to EIDs. The same applies to the best estimate analysis with quantification of uncertainties.

The recommendations related to the various assumptions for performing DSA for different plant states are applicable to water cooled SMRs.

The recommendations related to DSA documentation and independent verification are applicable to EIDs. However, owing to the limited scope of validation of computer codes for specific EIDs and the rather limited experience of code users in the analysis of areas of novelty in the design as described in Section 3 of this publication, it is challenging for practitioners to fully comply with the recommendations on independent verification.

The descriptions provided in annex I of SSG-2 (Rev. 1), except for those regarding application of DSA to the development of severe accident management guidelines, are written in a technology neutral manner and therefore are applicable to all reactor technologies.

The general concept provided in annex II of SSG-2 (Rev. 1) is applicable to EIDs. However, the specific frequency ranges might need to be revisited (see discussion in Section 4.11.2.3).

4.11.2.2. Areas of non-applicability

The following areas of non-applicability were identified:

(a) In section 3 of SSG-2 (Rev. 1) [89], the specific category of DECs with core melting is not relevant for some EIDs (e.g. for gas cooled reactors with TRISO fuel). This is because SSR-2/1 (Rev. 1) conflates severe accidents with occurrence of core melt.

(b) In section 7 of SSG-2 (Rev. 1), the recommendations on the analysis of core melt accidents are not applicable to HTGRs.
(c) The descriptions of the analysis in support of severe accident management guidelines provided in annex 1 of SSG-2 (Rev. 1) [89] are not applicable to HTGRs.

4.11.2.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) From a general point of view, SSG-2 (Rev. 1) [89] contains numerous examples that are specific to WCRs and that would have to be adapted or extended to make the Safety Guide more technology neutral.
(b) The specific design features of EIDs in terms of plant operating regimes may not be fully covered. The following three examples illustrate the issue:
    (i) For EIDs with long refuelling periods and factory refuelling, the identification of relevant refuelling modes is missing.
    (ii) For EIDs with very long refuelling periods, the risk during a refuelling period may be minimized at the plant and transferred, instead, to a centralized facility.
    (iii) For some EIDs operating at low pressure, power states and shutdown operating regimes may have a similar risk significance.
(c) The identification of PIEs does not consider specific features of EIDs, such as the configuration of modules, the lack of relevance of some SSCs used in conventional WCRs and the use of passive systems. TNPPs may be exposed to a range of faults that do not apply to traditional NPPs (e.g. transportation related).
(d) For technical acceptance criteria, the recommendations aimed at protecting the integrity of barriers are applicable (paras 4.12–4.15). However, appropriate selection of the criteria to reflect differences in barriers, associated phenomena and damaging mechanisms of barriers in EIDs are not included. In addition, additional guidance could be useful to address multi-modularity and multi-unit aspects.
(e) It would be useful to reflect the specifics of non-WCRs in providing examples on the functionality of the containment, in consideration of severe accidents (paras 6.5 and 6.6) and in giving examples of factors to be addressed in radiological analysis (para. 6.13).
(f) The recommendations on various assumptions that may be made when performing DSA for different plant states (section 7 of SSG-2 (Rev. 1)) may not cover the specificities of non-water cooled EIDs. In particular, confinement functions, elaboration on examples of relevant phenomena,

types of event to be analysed, applicable acceptance criteria and the single failure criterion may not be fully addressed. In addition, the discussion of an accident potentially leading to large radioactive releases does not include a definition of this term for NPPs including EIDs.

(g) The concept of frequency ranges of the plant states provided in annex II of SSG-2 (Rev. 1) [89] may be applicable to all EIDs. Nevertheless, for some EIDs, the frequency range for DBAs could fall below the ranges cited because of the extensive use of passive systems and inherent safety.

4.11.3. IAEA Safety Standards Series No. SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants

4.11.3.1. Areas of applicability

The general considerations provided in section 2 of SSG-3 [90] for Level 2 and Level 3 PSA metrics are applicable to all types of EID.

In addition, the recommendations in section 3 of SSG-3 [90] on project management and organization for PSA and in section 4 on familiarization with the plant and collection of information are found to be applicable to all types of EID, noting that, as for an NPP at early design stages, the recommendations in section 4 may not be fully implementable at the early stages of the design of EIDs.

Concerning the recommendations in section 5 of SSG-3 on initiating event analysis and accident sequence analysis, the general methodology is generally applicable to all reactor technologies (e.g. selection process, screening). Owing to the scarcity of operating experience and detailed design information, the use of information from similar EIDs will be initially limited and, thus, the main focus is expected to be on deductive and inductive analytical methods for identification of initiating events (see para. 5.13 of SSG-3).

The recommendations in section 5 of SSG-3 on system analysis, data analysis, analysis of dependent failures, human reliability analysis and CCF analyses are written in a technology neutral manner and therefore are generally applicable to all reactor technologies. The application of current human reliability analysis methods might not adequately address human failure events with long time windows, which is often the case for some EIDs.

The recommendations on quantification of accident sequences, and importance and sensitivity analyses are generally applicable to all reactor technologies, given the limitations mentioned regarding the applicability of the concept of core damage frequency (see discussion in Section 4.11.3.3).

The general methodology for Level 1 PSA for internal and external hazards is found to be applicable to all reactor technologies.

4.11.3.2. Areas of non-applicability

The following areas of non-applicability were identified:

(a) Review of section 2 of SSG-3 [90] indicates that the consideration of various PSA end states (e.g. core damage for Level 1 PSA) may not be applicable to all EID technologies (see paras 2.2–2.4 of SSG-3). The recommendations in SSG-3 imply calculation of the core damage frequency. Therefore, the PSA studies for EIDs that directly go to Level 3 end states (without quantifying Level 1, Level 2 or both end states) are not in line with SSG-3 in this respect. Nevertheless, the recommendations provided in SSG-3 could be used for supporting the development of PSA models that go directly to Level 3 end states (e.g. for most of the PSA elements, such as initiating events analysis, system reliability analysis, data analysis, CCF analysis, human reliability analysis). When referring to the safety goals and criteria in section 2 of SSG-3 (see paras 2.10–2.20), however, the discussion of the Level 1 metrics (i.e. core damage frequency) may not be applicable to EIDs for which core damage may not be meaningful, such as HTGRs.
(b) Section 9 of SSG-3 [90] is dedicated to the consideration of low power and shutdown modes and is generally applicable. However, it includes some recommendations that are more relevant to water cooled SMRs than to other types of EID (e.g. the parameters in para. 9.8 include references to the water level in the primary system). Efforts need to be made to identify more broadly applicable physical and technical aspects for the plant states of various non-water cooled EIDs.

4.11.3.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) The end states of accidents at some EIDs are different from the traditional core damage in WCRs (also large releases for Level 2 PSA; see discussion on SSG-4 [91] in Section 4.11.4). Therefore, current risk measures, such as core damage frequency, may not be meaningful to EIDs, and current guidance does not cover new risk approaches that may be needed for the risk informed decision making process.
(b) Certain additional guidance might be beneficial for specific technologies (e.g. HTGRs) in terms of new Level 1 end states considering the challenges for the barriers. It is conceivable that HTGRs may need multiple Level 1 end states corresponding to, for example, excessive radionuclide diffusion

through TRISO particles, excessive oxidation of structural components and excessive contaminated dust resuspension (i.e. for pebble bed designs).

(c) The general methodology for initiating event analysis and accident sequence analysis is applicable to all reactor technologies. However, EIDs also have PIEs that are different from those in WCRs, as discussed in Section 3. For example, for EIDs with modular design, current guidance does not address the identification of internal events in the context of multi-modularity (e.g. a group of modules sharing mitigating equipment). In addition, the unique hazards associated with a specific type of EID may not be addressed. This includes hazards associated with cleanup systems, as well as any high temperature chemical phenomena that may occur at the plant (e.g. sodium fires). A gap exists in identifying and evaluating the range of unique hazards or potential initiators that may exist for novel designs.
(d) The treatment of failure data generated from a technology development programme (see para. 5.123 of SSG-3) may not be covered. The approaches on how the technology development reliability data can be appropriately integrated into a PSA before sufficient operating experience is developed may not be addressed.
(e) EIDs are expected to rely on software reliability and passive system reliability, so modelling issues, including CCFs, may not be covered.

4.11.4. IAEA Safety Standards Series No. SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants

4.11.4.1. Areas of applicability

The general considerations provided in section 2 of SSG-4 [91] on project management and organization for PSA are found to be applicable to all types of EID. The recommendations in section 2 of SSG-4 may not be fully implementable at early stages of the design of EIDs.

The recommendations in sections 4 and 5 of SSG-4 on interfaces and accident progression analysis in Levels 1 and 2 of the PSA apply to water cooled SMRs. The same conclusion was reached for section 6 of SSG-4, which is dedicated to the source terms for severe accidents.

4.11.4.2. Areas of non-applicability

The following areas of non-applicability were identified:

(a) Some recommendations in section 3 of SSG-4 [91] (e.g. para. 3.2) imply a structural containment, and hence are not applicable to the functional containment approach used in HTGRs and other reactor concepts using TRISO fuel.
(b) Section 4 of SSG-4 [91] is dedicated to the interface between Level 1 and Level 2 PSAs and refers to the core damage concept, which is not applicable to all reactor types (see the corresponding discussion for SSG-3 [90] in Section 4.11.3.2). In addition, as highlighted above, the discussion on interfaces sometimes implies a structural containment and is not applicable for the 'functional containment' approach for specific EIDs. For instance, para. 4.7 of SSG-4 is not applicable to non-WCR designs.
(c) The recommendations in section 5 of SSG-4 [91] on accident progression analysis are not always applicable to non-WCRs. For instance, para. 5.1 of SSG-4 states that where the containment does not exist, the recommendations in section 5 are not entirely applicable. This means that most of that section is incompatible with a functional containment approach.
(d) The discussion provided in section 6 of SSG-4 on source terms for severe accidents is not applicable to non-WCRs. In particular, para. 6.12 of SSG-4 [91] is too restrictive for EIDs, for which thermal fluid behaviour and radionuclide release behaviour could be separately calculated. In addition, the core damage concept specified in paras 6.3 and 6.11 of SSG-4 [91] is not applicable to HGTRs.

4.11.4.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) The guidance in section 3 of SSG-4 does not support the consideration of design aspects that are specific to EIDs (e.g. associated with the areas of novelty provided in Section 3).
(b) The guidance does not address the severe accident conditions that could result if passive systems are extremely challenged.
(c) The guidance does not support EIDs, such as HTGRs, that may have releases to the atmosphere without an ex-vessel phase of a severe accident (see para. 6.13 of SSG-4). The guidance does not consider the retention of radionuclides in the coolant, which may be important in SFRs and LFRs.

(d) Gaps were identified regarding the consideration of hazards in shutdown mode for various EIDs. One example is dust resuspension during maintenance activities at HGTRs (especially for pebble bed designs).
(e) The guidance does not consider computational tools for a Level 2 PSA that might not have the same level of integration as the tools used for WCRs. For example, sodium reactors may calculate the thermal fluid evolution of the reactor in a suitable systems code, and then calculate evaporation of radionuclides from the sodium using a separate chemistry code using the systems code's temperature outputs as a boundary condition.

4.11.5. IAEA Safety Standards Series No. SSG-25, Periodic Safety Review for Nuclear Power Plants

4.11.5.1. Areas of applicability

SSG-25 [92] is applicable to EIDs.

4.11.5.2. Areas of non-applicability

No areas of non-applicability were identified.

4.11.5.3. Identified gaps and areas for additional consideration

The guide does not provide guidance on the following cases:

(a) The period of the safety review cycle (typically 10 years) may not align well with the periodicity of the refuelling cycle. For example, some EIDs being proposed have refuelling cycles of multiple decades or permanently fuelled cores (i.e. lifetime cores).
(b) An EID may originally be built with a few power modules, but subsequent modules may be added over time.
(c) For an EID, the entire reactor may be replaced on a periodic basis (could range from 4 years to more than 20 years, for example).
(d) For marine based TNPPs, a new 'power platform' may be brought in to replace an existing platform that is undergoing a refit. This is not covered in the approaches used for current PSRs.
(e) The development of a logical periodicity of the PSR based on the above considerations is not considered in the guidance.

4.11.6. IAEA Safety Standards Series No. NS-G-2.13, Evaluation of Seismic Safety for Existing Nuclear Installations

4.11.6.1. Areas of applicability

Safety Guide NS-G-2.13 [93] (under revision) provides acceptable methods for the seismic safety evaluation of existing nuclear installations. Such evaluations are normally prompted by a seismic hazard perceived to be greater than originally established for defining the seismic design basis. Seismic design and qualification are distinct from seismic safety evaluation. Seismic safety evaluation of existing installations strongly depends on the actual condition of the installation at the time the assessment is performed.

The scope of NS-G-2.13 is not limited to any particular reactor technology, but there are aspects that may be very specific to some technologies (e.g. the sloshing effect in LFRs and SFRs that may be more significant compared with WCRs) and which may require judgement.

Floating TNPPs are not directly subjected to seismic waves, since they are seismically isolated from the ground by the water. However, when moored at a particular location, seismic safety needs to be assessed for appurtenances, protective structures or nearby slopes, whose seismic failure might affect the safety of the reactor. These aspects are covered by NS-G-2.13.

4.11.6.2. Areas of non-applicability

No areas of non-applicability were identified.

4.11.6.3. Identified gaps and areas for additional consideration

Seismic induced water waves (e.g. seiche, tsunami) need to be considered for TNPPs. They are not in the scope of NS-G-2.13 but are covered by SSG-18 [27].

4.12. EMERGENCY PREPAREDNESS AND RESPONSE

4.12.1. IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear and Radiological Emergency

4.12.1.1. Areas of applicability

The safety requirements established in GSR Part 7 [14] apply for preparedness and response for any nuclear or radiological emergency that could occur in relation to a facility, an activity or a source, irrespective of the cause. They are written in a technology neutral manner and, thus, are applicable to any EID.

In its essence, GSR Part 7 [14] applies a graded approach on the basis of hazards associated with a facility, an activity or a source, as well as potential consequences of an emergency if it occurs (Requirement 4). In this context, it defines five EPCs, with EPCs I to III being associated with facilities, and EPC IV associated with activities involving mobile dangerous sources and unauthorized acts associated with them. It also requires that a hazard assessment is performed by Member States to inform decisions on the level of preparedness and response needed, and associated emergency arrangements. Hence, any decisions on what appropriate emergency arrangements would be for EIDs need to be based on a hazard assessment, provided that this hazard assessment considers the following:

(a) Events of very low probability and events not considered in the design, which include those triggered by a nuclear security event (e.g. sabotage);
(b) Events involving a combination of nuclear or radiological emergency with conventional emergency that could affect wider areas and/or could impair capabilities to provide support in the emergency response;
(c) Events affecting several facilities and activities concurrently, as well as interactions between them;
(d) Events at such facilities and activities elsewhere.

Thus, operators and Member States would need to perform such a hazard assessment and categorize an EID to a specific EPC. Then, the respective requirements of GSR Part 7 [14] that are applicable to that EPC would also be applicable to the EID. The annex of GSR Part 7 provides an overview of the applicability of paragraphs in GSR Part 7 by EPC. Such a hazard assessment will impose certain requirements on the operational arrangements, such as the characteristics of the emergency response facilities, their location, staffing, operational protocols with off-site emergency services, need for EPZs and emergency planning distances (EPDs), and comprehensiveness of emergency arrangements within them.

4.12.1.2. Areas of non-applicability

No areas of non-applicability were identified.

4.12.1.3. Identified gaps and areas for additional consideration

No gaps were identified.

4.12.2. IAEA Safety Standards Series No. GS-G-2.1, Arrangements for Preparedness for a Nuclear or Radiological Emergency

4.12.2.1. Areas of applicability

GS-G-2.1 [94] provides recommendations on meeting several safety requirements in the area of EPR. These recommendations are formulated in a technology neutral manner and on the basis of the EPC of a facility, an activity or a source. Several parts of this Safety Guide that are addressing specific facilities or activities in different EPCs are further analysed in this section in terms of their applicability for EIDs.

The guidance provided in this publication on the criteria for determining the EPC for facilities and activities derive from past experience and are given in terms of power levels, inventory of dispersible radioactive material and potential for uncontrolled criticality. Provided that a hazard assessment is performed to inform on the events involving EIDs that may lead to an emergency warranting protective and other response actions and associated hazards, these criteria can also be used to categorize EIDs to a specific EPC and allocate respective recommendations that are applicable to that EPC.

Section 6 of GS-G-2.1 [94] provides the concept of operations — namely a brief description of what the ideal response to different emergency scenarios is expected to look like for the purpose of ensuring that all those who are engaged in the emergency planning share a common understanding. The basic operations are generally applicable to any emergency scenario involving an EID in its specific EPC.

Similarly to the criteria for determining the EPC, the sizes of EPZs are suggested in appendix II of GS-G-2.1 on the basis of technologically neutral criteria. The appropriate size of these zones for EIDs can be identified using the criteria provided in terms of reactor power and inventory. The actual sizes of these areas are expected to be determined after thorough justification and optimization that takes into account a range of various factors (e.g. number of affected population, characteristics of the affected areas, resources available) regardless of whether technologies currently in use are considered or EIDs.

Appendices IV and VI describe emergency classes for facilities in EPCs I to III, as well as the response time objectives associated with activating critical functions for emergency associated with any of the EPCs I to V. As such, both appendices are technologically neutral, addressing various facilities in a general manner and, thus, are applicable to EIDs.

Functions and characteristics of emergency response facilities and locations are described in appendix VIII of GS-G-2.1. These functions and characteristics might have some specificities that are local and site specific or technology based when they are considered in relation to EIDs. However, as initially discussed, such details and specifics are usually not addressed at the level of a General Safety Guide. Still, the basic functions and characteristics of such facilities and locations are generally applicable to EIDs.

4.12.2.2. Areas of non-applicability

No areas of non-applicability were identified.

4.12.2.3. Identified gaps and areas for additional consideration

(a) Appendix I of GS-G-2.1 does not include EIDs. Their inclusion needs to be considered once EIDs can be characterized and assessed sufficiently and reliably.
(b) Areas requiring further clarification to be considered within GS-G-2.1 include: (i) determining the EPCs for EIDs, including those with several modules; (ii) determining the EPCs for transportable NPPs; and (iii) feasibility and appropriateness of mobile EPZs and EPDs for TNPPs.

4.12.3. IAEA Safety Standards Series No. GSG-2, Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency

4.12.3.1. Areas of applicability

GSG-2 [95] elaborates on the framework for emergency response criteria by providing generic criteria for taking protective and other response actions in a nuclear or radiological emergency. The Safety Guide also provides operational criteria to be used to initiate specific actions during the emergency response. As such, the guidance and recommendations provided in GSG-2 are applicable to EIDs. Appendix III of GSG-2, however, discusses the development of emergency action levels (specific to a facility) and provides an example of emergency action levels for light water reactors only.

4.12.3.2. Areas of non-applicability

No areas of non-applicability were identified.

4.12.3.3. Identified gaps and areas for additional consideration

An example of emergency action levels is given in appendix III of GSG-2 [95] for a light water reactor. The guide does not cover examples of emergency action levels for EIDs. Their inclusion may be considered once typical EIDs are characterized and assessed sufficiently and reliably.

4.12.4. IAEA Safety Standards Series No. GSG-11, Arrangements for the Termination of a Nuclear or Radiological Emergency

4.12.4.1. Areas of applicability

GSG-11 [96] addresses arrangements for the termination of a nuclear or radiological emergency in a manner that is facility and technology neutral and, as such, all the guidance and recommendations therein are applicable to EIDs.

4.12.4.2. Areas of non-applicability

No areas of non-applicability were identified.

4.12.4.3. Identified gaps and areas for additional consideration

No gaps were identified.

4.12.5. IAEA Safety Standards Series No. GSG-14, Arrangements for Public Communication in Preparedness and Response for a Nuclear or Radiological Emergency

4.12.5.1. Areas of applicability

GSG-14 [97] addresses arrangements for public communication in preparedness and response for a nuclear or radiological emergency in a manner that is facility and technology neutral and, as such, all the guidance and recommendations therein are applicable to EIDs.

4.12.5.2. Areas of non-applicability

No areas of non-applicability were identified.

4.12.5.3. Identified gaps and areas for additional consideration

No gaps were identified.

4.12.6. IAEA Safety Standards Series No. SSG-65, Preparedness and Response for a Nuclear or Radiological Emergency Involving the Transport of Radioactive Material

4.12.6.1. Areas of applicability

SSG-65 [98] addresses emergency preparedness and response for the transport of radioactive material, irrespective of the initiator of the emergency, which could be a natural event, a human error, a mechanical or other failure, or a nuclear security event. It elaborates on emergency preparedness and response arrangements for four different modes of transport: by road, rail, sea inland waterway and air. The scope of this Safety Guide is limited to transport involving nuclear or radioactive material under EPC IV and excludes emergency preparedness and response for reactors used to provide power for the propulsion of vessels.

4.12.6.2. Areas of non-applicability

No areas of non-applicability were identified.

4.12.6.3. Identified gaps and areas for additional consideration

No gaps were identified. An area that requires further clarification concerns the emergency arrangements and associated EPC that would be needed during transport of land based TNPP components loaded with fuel.

4.13. LEGAL AND REGULATORY FRAMEWORK

4.13.1. IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety

4.13.1.1. Areas of applicability

All requirements in GSR Part 1 (Rev. 1) [8] are applicable to EID technology and deployment.

4.13.1.2. Areas of non-applicability

No areas of non-applicability were identified.

4.13.1.3. Identified gaps and areas for additional consideration

Some EID deployment models require a close interaction and collaboration of regulatory bodies in different jurisdictions. According to the safety standards, an EID fuelled in a foreign factory, for instance, must be licensed to operate by the regulator in the vendor country and by the regulator in the receiving country. These two regulators, while reaching decisions independently, ought to work very closely to increase efficiency and minimize duplication. In light of these considerations, the following gaps were identified:

(a) Guidance is not included on effective cooperation, assistance and sharing of experience between regulatory bodies to guarantee safety and protection of the population and to use resources efficiently.
(b) Guidance is not available on transfer or sharing of oversight from one regulatory body to another, and on potential sharing of regulatory responsibilities for the different stages of the lifetime of the facility.

4.13.2. IAEA Safety Standards Series No. GSG-12, Organization, Management and Staffing of the Regulatory Body for Safety

4.13.2.1. Areas of applicability

All recommendations in GSG-12 [16] are applicable to EID technology and deployment.

4.13.2.2. Areas of non-applicability

No specific areas of non-applicability were identified.

4.13.2.3. Identified gaps and areas for additional consideration

No guidance is provided on regulatory oversight of many (similar) small reactors scattered over a country and in remote areas (possibly with limited infrastructure).

4.13.3. IAEA Safety Standards Series No. GSG-13, Functions and Processes of the Regulatory Body for Safety

GSG-13 [17] partially supersedes SSG-12 [99] and needs to be read in conjunction with SSG-12 and with GSG-12 [16].

4.13.3.1. Areas of applicability

All recommendations in GSG-13 [17] are applicable to EID technology and deployment.

4.13.3.2. Areas of non-applicability

No areas of non-applicability were identified.

4.13.3.3. Identified gaps and areas for additional consideration

The guide lacks explicit guidance on the need for early engagement of the regulator(s) by the designers when developing new designs. This may be more important for EIDs than for current WCRs owing to the areas of novelty of EIDs.

4.13.4. IAEA Safety Standards Series No. SSG-12, Licensing Process for Nuclear Installations

SSG-12 [99] was partially superseded by GSG-13 [17] and needs to be read in conjunction with GSG-13 and with GSG-12 [16].

4.13.4.1. Areas of applicability

All recommendations in SSG-12 are applicable to EID technology and deployment.

4.13.4.2. Areas of non-applicability

No areas of non-applicability were identified.

4.13.4.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) The areas of novelty of EIDs may require more interactions between the designer and the regulator to compensate for the regulator's lack of familiarity with the design. These early interactions/engagement will help the regulator in the review process. Guidance on this is currently missing from SSG-12 [99].

(b) Some types of EID are more likely to be manufactured in a foreign factory or may be refuelled abroad. Such activities must be overseen by regulators from both the vendor and the receiver countries. These two regulators, while reaching decisions independently, ought to work very closely to increase efficiency and minimize duplication. This will be of particular value where the technology is new. There may also be potential for sharing of regulatory responsibilities. These issues are currently not covered by SSG-12.

(c) The Safety Guide lacks explicit guidance on the need for early engagement of the regulator(s) by the designers when developing new designs. This is more relevant for EIDs because of their areas of novelty and lack of operating experience.

(d) Although the described generic process may be applied to all models of deployment of EIDs, SSG-12 may lack some potential process details that would apply to some EID deployment models.

4.13.5. IAEA Safety Standards Series No. GSG-6, Communication and Consultation with Interested Parties by the Regulatory Body

4.13.5.1. Areas of applicability

All recommendations in GSG-6 [15] are applicable to EID technology and deployment.

4.13.5.2. Areas of non-applicability

No areas of non-applicability were identified.

4.13.5.3. Identified gaps and areas for additional consideration

No gaps were identified.

4.13.6. IAEA Safety Standards Series No. SSG-16 (Rev. 1), Establishing the Safety Infrastructure for a Nuclear Power Programme

4.13.6.1. Areas of applicability

SSG-16 (Rev. 1) [100] addresses all aspects of the necessary safety infrastructure. It is intended for use by persons or organizations participating in the preparation and implementation of a nuclear power programme. Uniquely, SSG-16 (Rev. 1) contains ‘actions’, which are overarching recommendations derived from the referenced safety requirements. The explanations that accompany these actions are mainly based on the referenced safety requirements and associated Safety Guides. Therefore, SSG-16 is to be read in conjunction with all referenced IAEA safety standards.

Overall, the recommendations set out in SSG-16 (Rev. 1) are considered generally applicable to EIDs technology and deployment.

4.13.6.2. Areas of non-applicability

No areas of non-applicability were identified.

4.13.6.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) Because the actions provided in SSG-16 (Rev. 1) are derived from the referenced safety requirements, if there is an identified gap or area for additional consideration for EIDs in a referenced specific safety requirement or guide, this could be considered as a gap or area for additional consideration in SSG-16 (Rev. 1) regarding its applicability to EIDs.
(b) Although most of the actions provided in SSG-16 (Rev. 1) are generally applicable to EIDs, some actions and explanatory texts regarding participating in the construction; overseeing the activities performed by the contractors, particularly the manufacturing; siting; preparation for commissioning; transport safety; and radioactive waste management, spent fuel management and decommissioning could need some revisions to cover all types of EID and deployment model.

4.14. TRANSPORT OF RADIOACTIVE MATERIAL

4.14.1. IAEA Safety Standards Series No. SSR-6 (Rev. 1), Regulations for the Safe Transport of Radioactive Material

4.14.1.1. Areas of applicability

The most recent IAEA regulations for the safe transport of radioactive material, provided in SSR-6 (Rev. 1) [23], are applicable to the transportation of EID fresh nuclear fuel or SNF by surface or sea (described as Scenario 1 in Section 3.9.2). These regulations are implemented worldwide through regional agreements and/or national legislations for surface transport and through the IMDG Code [24] for marine based transport, which would apply to all EIDs. For these cases, SSR-6 (Rev. 1) requires that fresh fuel and spent fuel are transported in an appropriate package type (for fissile packages) as approved by the competent authority.

If a TNPP contains used fuel in a package and is transported by sea, the ship is also subject to the IMO International Code for the Safe Carriage of Packaged Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Wastes on Board Ships [130]. Further, within the current IMO regulatory framework, there is a 'Nuclear Ship Category' within the International Convention for the Safety of Life at Sea (SOLAS) [131], which applies to ships that have a nuclear propulsion system.

4.14.1.2. Areas of non-applicability

The following areas of non-applicability were identified:

(a) For TNPPs containing fresh nuclear fuel or SNF (Scenario 2 in Section 3), transport using the types of package specified in SSR-6 (Rev. 1) [23] may not be possible, particularly for large size TNPPs.
(b) The scope of SSR-6 (Rev. 1) [23] does not include the design of ships that carry nuclear reactors (TNPPs for transport by sea); neither are these covered by SOLAS [131], unless they have a nuclear propulsion system.

4.14.1.3. Identified gaps and areas for additional consideration

The following gaps and areas for additional consideration were identified:

(a) For Scenario 1, as defined in Section 3.9.2, a TNPP is always transported empty of fuel. This might allow a previously operational TNPP to be shipped as a surface contaminated object or as low specific activity material. There is, however, no experience on whether the relevant limits would be met or whether a special arrangement, as specified in SSR-6 (Rev. 1), would be necessary.

(b) Scenario 2, as defined in Section 3.9.2, envisages transport of a TNPP including fresh or used fuel by road, rail or water. This situation is not explicitly covered by SSR-6 (Rev. 1) [23]. Furthermore, the regulatory process associated to the potential transport of EIDs with fresh nuclear fuel and SNF has not yet been fully developed by States in cooperation with the IAEA, and there is currently no mechanism to enable multilateral approval of the transport by the relevant competent authorities other than SSR-6 (Rev. 1) [23]. As noted above, transport by sea of a TNPP containing used fuel also falls outside the scope of SOLAS [131].

5. SAFETY, SECURITY AND SAFEGUARDS INTERFACES FOR EIDs

An integrated approach to safety, security and safeguards is especially essential in the context of rapid development of EIDs. With many EIDs still in the conceptual design stage, designers have the opportunity to consider safety, security and safeguards at an early stage and to do this in a way that makes use of potential synergies and avoids possible conflicts between the three: the so-called 3S concept (see Section 2.5.1).

This section evaluates the areas of novelty identified in Section 3 from the security and safeguards perspective and considering their mutual interfaces and their interfaces with safety, in the spirit of the 3S concept. Specific areas of novelty that are assessed in this section include transportability, new fuel concepts, limited access and remote locations, non-electrical applications and the possibility for increased potential for cyberattacks.

This section provides a description of potential challenges for the application of the current practices on security and safeguards described in Section 2.4 to EIDs, which relate to the areas of novelty of EIDs and are identified as important from the security, safeguards and 3S perspective. These challenges are described in the subsections entitled ‘Identified areas for additional consideration’. This description is aimed at providing general information on potential challenges and is considered to be a starting point for further development of guidance on the 3S concept for EIDs. The detailed analysis of potential gaps in IAEA publications in the area of safeguards and security gaps is out of scope of this Safety Report.

For the consideration of the 3S concept, this section presents a systematic approach to safety, security and safeguards through the description of potential interfaces and any resulting conflicts and synergies. Thorough understanding of the interfaces is an important prerequisite for setting up the relevant measures (e.g. design solutions, organizational matters, regulatory framework) for effective integration of these three disciplines for EIDs and for avoiding or minimizing potential negative interactions.

5.1. SAFEGUARDS CONSIDERATIONS

This section presents the assessment, from a safeguards point of view, of EID areas of novelty and their potential impacts on the application of existing IAEA safeguards approaches to EIDs described in Section 2.4. On the basis of the assessment, areas for additional consideration are identified.

The section is organized by topical area (e.g. transportability, fuel concepts, remote locations, non-electrical applications). For each topic, the area of novelty and why it exists is briefly explained. A summary table of safeguards challenges for different types of EID is provided in Appendix I. It is recognized that some safeguards measures that are applied to other nuclear installations (e.g. on-line refuelled power reactors, research reactors, uranium enrichment facilities, fuel reprocessing facilities) may be adapted to EIDs.

There may be overlap between the topics discussed in the sections below, as some of them impact multiple reactor design features. This is pointed out to the extent possible in this brief analysis.

5.1.1. Transportability and modularity of reactors with the fuel loaded

As identified in Section 3, some EIDs (mainly SMRs) may be smaller and be deployed in large numbers and therefore be more numerous than conventional WCRs. Some EIDs are foreseen to be entirely manufactured at a central factory, which might imply that the main modules of the system, including reactors loaded with fuel, are fabricated at a factory and then assembled on-site. Some microreactors are also often conceived as fully fabricated at a factory and then transported to the site of operation with the fuel already in place. Such schemes eliminate fuel loading/unloading activities at the site of deployment. From the point of view of safeguards there are two main possibilities:

(a) Transport of the TNPP to the site of installation with the reactor empty of fuel. Fuel is loaded at the site, and later the TNPP empty of fuel, but still containing radioactive material, is transported to another site or back to the manufacturer (referred to as Scenario 1 in Section 3).

(b) Transport of the TNPP to the site of installation with fresh nuclear fuel already present in the reactor, and later the TNPP is transported to another site or back to the manufacturer, complete with irradiated fuel (referred to as Scenario 2 in Section 3).

Under a comprehensive safeguards agreement, IAEA safeguards are applied to all sources of special fissionable material in all peaceful nuclear activities. This is done for the exclusive purpose of verifying that such material is not diverted to nuclear weapons or other nuclear explosive devices.

Case (a) above corresponds to arrangements for existing reactors (see Section 5.1.2) from which it follows that safeguards arrangements would not need to change unless in response to unusual properties of the fuel itself.

Case (b) implies that the safeguards measures are to be applied at the central factory, during transport of the reactor and during its installation at the deployment site. The reactor factory and the fuel factories (fresh fuel factory and spent fuel reprocessing facility) may reside in a Member State different from the one in which the system is deployed, and such State could be either a non-nuclear weapon State or a nuclear weapon State [132][31]. In the case in which the State hosting the reactor factory and the fuel factories is not a non-nuclear-weapon State with a comprehensive safeguards agreement in force, adequate safeguards provisions need to be arranged in advance for ensuring that the reactor factory and the fuel factories are effectively safeguarded and the appropriate verification measures for design information are carried out.

5.1.1.1. Identified areas for additional consideration

While in transit, continuity of information needs to be maintained on factory-loaded reactors. In principle, the same safeguards provisions adopted for the transport of nuclear fuel also apply to the transport of nuclear reactors.

[31] Signatory States of the Treaty on the Non-Proliferation of Nuclear Weapons adhere to the treaty as either a non-nuclear weapon State or a nuclear weapon State. Article IX of the Treaty on the Non-Proliferation of Nuclear Weapons defines a nuclear weapon State as a State that "has manufactured and exploded a nuclear weapon or other nuclear explosive device prior to 1 January 1967."

Compared with the transport of nuclear fuel, the transport of an entire reactor adds additional sensitivities:

(a) A fully loaded core implies the availability of several significant quantities of special fissionable material in case of diversion.
(b) Specific reactor designs (e.g. TNPPs) are attractive for diversion, as they provide the diverter with not only the nuclear material to be irradiated, but also with the facility itself, enabling potential misuse of the technology.

In addition to the above considerations, the special case of TNPPs that are foreseen to be transportable throughout their operational life cycle (e.g. floating TNPPs) adds an additional layer of complexity that touches inter alia the following aspects:

(a) The definition of site: The current definition of site, as given in Ref. [109], assumes it to be a geographically precise static area containing the nuclear facility, definable from the design information of the facility itself. In contrast to typical nuclear installations, transportable nuclear reactors are intrinsically mobile, and no site could be identified from their design information. In addition, special provisions are to be considered not only for the continuity of knowledge of the usual parameters, but also of the location of the site and related material balance areas (MBAs).
(b) The validity of safeguards provisions beyond national geographical limits: A transportable nuclear reactor could, in principle, be moved outside the borders of the State 'owning' it to either other States with different safeguards agreements or to extraterritorial areas (e.g. the high seas). Special provisions need to be put in place to ensure the applicability of IAEA safeguards verifications in any potential circumstance, from both a legal point of view (adequate arrangements) and a practical point of view (accessibility, availability of all the boundary conditions needed to incorporate IAEA safeguards measures).
(c) Difficulty in accessing the mobile or floating reactor: This may be due to the special configuration and location of the facility. In this case, the IAEA does not have any experience with performing inspections and design information verification at such facilities during their transportation.

5.1.2. Transport of nuclear fuel

Transport of both fresh and irradiated nuclear fuel around the globe is routinely carried out since the beginning of the nuclear power industry, and effective safeguards provisions are already available and implemented routinely.

The future foreseen deployment of several small reactors or microreactors in very remote areas or, on the other end of the spectrum, in densely populated areas, such as industrial areas or within city boundaries, might bring an additional layer of complexity in safeguards implementation.

5.1.2.1. Identified areas for additional consideration

The following areas for additional consideration were identified:

(a) The ability of current safeguards techniques to guarantee adequate continuity of knowledge. Current practice foresees the application of IAEA containment and surveillance measures to the nuclear material being transported, such as an IAEA seal that verifies the integrity of the nuclear material's containment. A significant increase of nuclear fuel shipments in remote locations might challenge current safeguards approaches and result in the need for operators to apply or remove seals and/or for continuity of knowledge to depend on advanced surveillance techniques rather than on containment alone.
(b) The compatibility of current safeguards and security provisions with the novel potential siting and logistical arrangements of small reactors or microreactors[32]. The flexibility of these reactors in terms of siting and transportability could challenge existing safeguards and security procedures.

5.1.3. Small size of fuel items

SMRs tend to foresee fuels of smaller size than those of large WCRs. Microreactors can have either a single fuel item that acts as a 'cartridge' or several small fuel items that could, in principle, be replaced independently[33]. Irrespective of the individual design choices, smaller fuel items often imply less special fissionable material per assembly, with a need for diverting more items to acquire a significant quantity of special fissionable material. It is safe to assume that the smaller size of the fuel items would not impact significantly the technical difficulty of diverting them, as the fuel items will still need the use of

[32] In certain countries, the transport of certain types of special fissionable material needs the escorted transport truck to remain in motion until the end of the journey. To meet this challenge, the roads involved in the itinerary are closed to traffic until the convoy has transited. In highly densely populated areas, this might have unmanageable repercussions on the area's activities and might require the development of different transport provisions, which in turn might affect the way in which continuity of knowledge for safeguards purposes is maintained.

[33] The case of liquid fuel and pebble bed reactors will be treated in a separate section, in conjunction with the issues related to on-line refuelling.

special transport equipment. The need to divert a higher number of items in order to acquire a significant quantity of special fissionable material, coupled with a smaller core foreseeing a smaller number of fuel items per core, would likely make the detectability of such an action higher than in power reactors employing large fuel items.

5.1.3.1. Identified areas for additional consideration

The small size of fuel items and the smaller power density of some EIDs could open the door to space efficient layouts of spent fuel storages, which might foresee solutions such as stacking of nuclear fuel. This could create a potential safeguards challenge (and therefore an R&D opportunity) owing to less than acceptable accessibility to the nuclear material items for verification purposes and potential challenges in terms of ability to verify the attributes of SNF.

5.1.4. New fuels and fuel cycles

EIDs, particularly non-WCRs, may use new types of fuel composition and closed back end options. New types of fuel might lead to significant changes in current safeguards practices connected with the geometrical characteristics of the fuel (see Section 5.1.3), level of enrichment, additional fuel fabrication facilities, etc.

5.1.4.1. Identified areas for additional consideration

Some aspects of these novelties are not captured in IAEA publications and might require additional consideration, for example the following:

(a) Some EID designs aim to dispose of minor actinides by transmutation and, to this end, call for them to be mixed into fresh fuel. While the strong gamma radiation emitted by minor actinides might make it more difficult to divert such fuel and, in the process, enhance proliferation resistance, the higher radiation field may also present an accessibility barrier to verification. Moreover, the presence of minor actinides in new fuel might hinder the non-destructive analysis (NDA) conducted during safeguards inspections. It is possible that some aspects of spent fuel verification could be applied to circumvent such difficulties but the idea that attribute verification (the means of spent fuel verification) might be used in place of nuclear material characterization would need careful consideration.
(b) Some cores are foreseen to operate with very long irradiation cycles, and therefore potentially exhibit a higher than usual excess reactivity at the

beginning of the irradiation cycle. Where cores are not factory-loaded, this excess reactivity might provide opportunities for potential misuse, such as the undeclared irradiation of fertile nuclear material targets to produce weapons-usable nuclear material. This issue is magnified for fast spectrum EIDs, where the neutron losses are relatively high and can be used (in a declared or undeclared manner) to breed fissile materials. Moreover, longer irradiation cycles imply that an important part of the special fissionable material inventory will not be available for verification for long periods of time, requiring robust means to ensure continuity of knowledge.

(c) Some EIDs may operate as part of a closed nuclear fuel cycle. Consequently, large plutonium inventories will have to be verified and, possibly, characterized on-site and/or at fuel reprocessing facilities. In the case of MSRs, potentially large quantities of fuel salt will have to be verified and accounted for.

(d) Closing the nuclear fuel cycle implies the need to deploy recycling and reprocessing technologies, and many innovative nuclear systems are investigating advanced reprocessing techniques that depart substantially from the PUREX technology that is widely adopted today. On one hand, these novel techniques will often exhibit a higher intrinsic proliferation resistance compared with PUREX, but on the other hand, most of the current verification techniques applied to current aqueous reprocessing facilities would not be applicable to the new processes.

(e) Some EIDs may use HALEU to increase the lifetime of the core. This would require an increase in enrichment capacity to sustain the overall demand and would potentially increase the sensitivity of the nuclear material to be shipped. In the case of liquid fuel MSRs, the fuel concentration needs to be measured during operation in order to compensate for the fuel burnup. Thus, new safeguards measures might be needed for this type of reactor.

(f) Some EIDs may utilize the thorium fuel cycle, which is based on the utilization of fertile thorium to breed fissile ^{233}U to maintain reactivity. The main precursor of ^{233}U is ^{233}Pa, which has a half-life of 27 days. Appropriate NDA techniques, reference standards, codes and simulations, and additional types of verification techniques may need to be developed. This issue is particularly important for solid fuel concepts that require reprocessing of the fertile material to extract ^{233}U. In some liquid fuel MSRs, fertile thorium is directly introduced in the molten salt of the core without the need for separation of ^{233}U.

5.1.5. Limited access

Many EIDs will present situations in which the nuclear material inventory will reside in difficult to access areas. Possible reasons for this design choice are, for example, a factory-loaded core that does not have the need for on-site fuel manipulation, long irradiation cycles between refuelling outages, health and safety hazards due to very high radiation levels, such as in certain areas of MSRs, and no-access areas under inert atmosphere, as in certain liquid metal cooled reactor designs.

5.1.5.1. Identified areas for additional consideration

Having a substantial amount of the special fissionable material in difficult to access or inaccessible areas presents both advantages and limitations that need further consideration, as follows:

(a) Difficult to access nuclear material inventory increases the theoretical intrinsic proliferation resistance of a nuclear system, as any diversion action would be technically more difficult. In principle, a factory-loaded core, with no possible access to the core without breaching the reactor's integrity and no possibility of on-site fuel manipulation, might make a number of potential acquisition pathways implausible without prompt detection.
(b) When a difficult to access nuclear material inventory is static, safeguards measures can rely effectively and efficiently on robust (dual) containment and surveillance techniques for ensuring continuity of knowledge on the inventory.
(c) If the difficult to access nuclear material inventory is not static (e.g. in highly automated and remotely operated systems), there is a need for complex and advanced monitoring and surveillance techniques, and current practice might not be suitable for the purpose.
(d) Irrespective of their robustness, containment and surveillance systems can fail. Hence, proper and effective reverification strategies and techniques need to be developed to ensure effective restoration of the lost continuity of knowledge.
(e) Finally, EIDs with a difficult to access nuclear material inventory might also present challenges in terms of design inventory verification: some EIDs are conceived to be operated in a highly automated way, with very limited need for human presence on the site. This might lead to design choices that make most of the system areas completely inaccessible to inspectors, requiring development of alternative techniques to verify the system design.

5.1.6. Remote locations and autonomous operations

Some EIDs are designed to operate autonomously or semi-autonomously with limited or no staff and in remote locations.

5.1.6.1. Identified areas for additional consideration

Access by IAEA inspectors might be limited in these cases. There are two separate but interrelated issues that need further consideration:

(a) The reactor is located remotely and has limited staff. In this case, access to the location will be similar to that of other nuclear fuel cycle facilities operating in remote areas, such as those for uranium mining and milling (although mines and mills are not subject to IAEA safeguards inspections under a comprehensive safeguards agreement but to complementary access under the Additional Protocol). The challenge is that IAEA inspectors may need long lead times to plan for access to these locations. This limits the possibility of having short notice or unannounced inspections and increases both the cost of conducting verification activities on-site and the burden placed on the operator. Other established cost effective techniques, such as viewing camera footage during inspections, would also be more difficult. One obvious solution is to employ existing or new remote verification techniques for which safeguards relevant information is authenticated and received at IAEA Headquarters. These signals could originate from IAEA specific instrumentation or shared equipment with the operator. Since these types of EID would typically be small (e.g. microreactors), they may be deployed in multiple locations so that one approach can be duplicated many times; this will reduce the cost of development and deployment. Additional challenges could arise if reliable on-site IT infrastructure does not exist at remote locations.

(b) The reactor is operated autonomously or semi-autonomously. Some EIDs are meant to work almost like nuclear batteries where they are installed, connected to the end user(s), and then operate autonomously with little or no operating staff. This situation has not been encountered before for operating NPPs, regardless of their location. If the NPP is in a remote location, this issue would be magnified, since access to the site would presumably require operator staff to be present (normal practice during on-site work by IAEA inspectors). Otherwise, assuming that the State would allow inspectors unaccompanied access to a nuclear facility, the IAEA might utilize new verification techniques. The potential for remote verification techniques described in (a) would also be applicable.

5.1.7. Non-electrical end use and intermittent operations

For some EIDs, the primary objective is not to provide continuous power for an electrical grid, but to perform other roles such as district and industrial heat applications, hydrogen generation or backup generation for renewables.

The additional challenge of this kind of intermittent operation for the IAEA is to verify that: (a) nuclear material has not been moved during non-operating periods, and (b) the increased potential for short cycling and production of plutonium from uranium targets. In this case, the IAEA experience with research reactors is relevant, since they are typically operated intermittently when needed for research activities (e.g. testing of nuclear fuel or materials), for commercial purposes (e.g. radioisotope production, neutron transmutation doping of silicon) or for education and training. Comparing operating history with declared nuclear material inventories may increase the IAEA's confidence in the correctness and completeness of a State's reporting. For power reactors, this approach may introduce some complexities, such as the coupling of operations with secondary parties (industrial customers) where the IAEA's verification activities may have a negative economic impact for those industrial users. The principal means of determining the schedule of reactor operations is monitoring the power produced as the IAEA does currently by inspecting operation logs or other indicators.

5.1.7.1. Identified areas for additional consideration

There are no areas for additional consideration in relation to this topic. Current approaches can, in principle, be applied to EIDs. If the EID output is chiefly electricity, then the safeguards approach is relatively straightforward. If the output is chiefly heat (such as steam or hot gases), the safeguards approach may be slightly more complex than for current power reactors (WCRs, PHWRs) because the power output of the NPP may not be coupled directly to an electricity producing turbine generator but to some other process. However, the underlying principle of reactor power monitoring would remain the same.

5.1.8. Moving-fuel reactors and continuous on-line refuelling

Some EIDs employ unique methods of on-line fuelling that utilize fuel that is not fixed inside the reactor, but circulates through the reactor. Such designs include PBRs, where solid fuel spheres are continuously fed and withdrawn from the reactor core every few minutes. They also include liquid fuelled MSRs, where the fuel is dispersed in the salt matrix and may circulate rapidly through the reactor via loops and out to heat exchangers. Such fuel concepts pose specific challenges from safeguards perspectives, which are described in Section 5.1.8.2.

5.1.8.1. Pebble bed reactors

The core of a PBR could consist of numerous fuel element spheres (‘pebbles’) with small diameter (e.g. 6 cm), none of which is individually identified. These fuel pebbles descend slowly through the core under gravity and then, depending on burnup (which is individually measured for each pebble), are either returned to the top of the core or discharged for storage as spent fuel. Typically, a few thousand fuel pebbles are examined each day and they are cycled through the reactor three times to give a total in-reactor fuel residence time of around three years.

The safeguards challenges for PBRs will mostly revolve around accounting for the pebbles. The number of pebbles that will potentially need to be tracked at any given moment could be in the millions. This figure includes fuel pebbles that come in as fresh fuel, those that are circulating in the reactor system and those that have been removed from the reactor because of either damage or burnup. Additionally, some of the PBR designs use graphite spheres for moderation during operation, and some use varying enrichment levels, which means that, at any given moment, there are spheres of significantly different enrichment and burnup in the core.

HTGR fuel pebbles typically contain about 5 g of uranium (perhaps 0.5 g of fissile material) so a few thousand would be needed to make up a significant quantity. This makes diversion difficult. Additionally, processing of the spent fuel spheres has not been done on a commercial scale, and there are indications that such processing is much more difficult than current aqueous reprocessing, which aids non-proliferation.

The IAEA does have some experience in safeguarding PBRs. Germany operated two PBRs, the AVR and THTR-300, between the 1960s and the 1990s (THTR-300 used thorium as fertile material). These reactors were subject to comprehensive safeguards agreements. In addition, the HTR-PM in China is currently under IAEA safeguards as part of a voluntary offer agreement [133]. This is a unit that consists of two 250 MW(th) reactors that together drive a single steam turbine with an electrical output of 210 MW(e) using fuel enriched at 4.2% and 8.5% ^{235}U, respectively.

In order to detect credible diversion pathways, the main technical objectives of safeguards at PBRs would be as follows:

(a) Detect diversion of fresh fuel pebble containers or pebbles, with or without substitution by dummy fuels prior to loading to the reactor;
(b) Detect diversion of core fuels or irradiated target materials, with or without substitution by dummy fuels and falsification of operating records;

(c) Detect diversion of spent fuel pebbles with or without substitution by dummy fuels.

Fresh fuel pebbles containing LEU can be verified and accounted for prior to being fed into the reactor system. The drums that contain the pebbles can be accounted for, their tags checked and some drums randomly selected for NDA to verify the existence of ^{235}U. For the verification of pebbles, drums can be randomly selected, pebbles accounted for and at least one pebble per selected drum can be withdrawn for NDA (issues such as the homogeneity of the pebbles, as well as the number of random samples considering the population, would need to be investigated further). Spectrometry can be applied to verify the enrichment of ^{235}U. Surveillance measures could be applied to the loading area to monitor the fresh fuel pebble loading process.

Once equilibrium (or steady state) operations of a PBR are achieved, the input and output of pebbles is kept in dynamic balance. Therefore, the average discharge burnup of a spent fuel can be directly determined by the integral of power history and the refuelling speed. Consequently, the operational parameters, including integral power history and the refuelling data, are the important safeguards parameters for nuclear material accounting.

Once the reactor core is installed in the reactor containment building and starts its operation, it may not be accessible for the whole operational lifetime. After loading, the fuel pebbles will likely be verified only when they are discharged as spent fuel. To maintain a continuity of knowledge of fuel pebbles loaded to the reactor, NDA (with radiation detectors) and video surveillance could be installed at places where irradiated fuel pebbles could be removed from the reactor.

The spent fuel storage can have a containment and surveillance system, since it will likely be difficult to remeasure spent fuel pebbles stored in casks and silos. NDA systems can be used to verify that the pebbles loaded into the spent fuel cask are irradiated nuclear material. In addition, NDA systems to confirm the level of filling of spent fuel pebbles into the cask could be used. NDA equipment for registering the movement of any material emitting gamma radiation or neutrons can be installed at entry and exit points to complement the containment and surveillance systems. Ideally, there will be technical means in place to re-establish the continuity of knowledge if the applied containment and surveillance system fail. Since the reactivity can be correlated to the number of pebbles in the core, modelling tools could be developed, whereby the reactor physics characteristics could potentially support the IAEA in comparing power production records with fuel usage.

The pebble handling and counting systems will be crucial to determining material balances in the reactor core and its systems. Therefore, if IAEA safeguards are applied, the IAEA would likely need access to accounting data, equipment and locations associated with the automated pebble handling and waste discharge systems.

5.1.8.2. Liquid fuelled molten salt reactors

A liquid fuelled MSR has no solid fuel; instead, fissile, fissionable and possibly fertile material is dissolved in molten salt, which serves as a carrier for the fuel and as a coolant. The fuel circulates between the core, where heat is gained, and heat exchangers, where the heat is released. For fast reactor designs, reactivity in the core is controlled by geometry and for thermal reactor designs, by geometry and the presence of a moderator. The use of liquid fuel allows volatile fission products to escape and provides the possibility that the fuel may be continuously processed during operation (e.g. adding fuel and reprocessing the fuel). The MSR concept is very flexible: designs differ significantly in the types of fissile material, flux spectrum and breeding or burning ratios. For the purposes of this publication, on-line processing is assumed.

MSRs will likely be treated as 'bulk' nuclear material facilities for the application of IAEA safeguards. The nuclear material will always be in bulk form, including the fresh fuel that will likely be synthesized and delivered as frozen (solid) salt, the active inventory in the core, all the waste and processing streams and the final spent fuel — even though, since the fuel can in theory be processed indefinitely, there may not be any SNF until decommissioning, and even then the fuel could be sent to another MSR for continued use. The salt may be containerized when it arrives as fresh fuel and when it is placed in spent fuel containers; in both cases, it is assumed that the containers will be properly verified and sealed. Some small size MSR designs are designed to operate without fuel processing and with lifetime cores. In such cases, all fission products are retained in the reactor, and the reactor is removed when the fissile material is spent.

Material control and accountancy will be challenging for an MSR because there is currently no safeguards ready method to measure the nuclear material directly in the irradiated salt. Further, developing a measurement system would be challenging if a sampling method were to be used, because of the potentially high concentrations of nuclear material in the fuel salt and the attendant high radiation fields for short cooled fuel salt.

The IAEA has not had the opportunity to implement safeguards on MSRs, since the only operational example was the Molten Salt Reactor Experiment test reactor that was operated in the United States of America (USA) in the 1960s. The closest analogue for how safeguards might be applied to a liquid fuelled MSR may be safeguards at reprocessing facilities and, specifically, for pyroprocessing, since it uses high temperature molten salt solutions containing irradiated nuclear materials. While some R&D has been undertaken on pyroprocessing safeguards, IAEA safeguards experience is limited, with much of the experience coming from the Republic of Korea. Some of the key areas in an MSR where safeguards might be applied include the following:

(a) Fuel salt synthesis (manufacturing): It is possible that some designs will synthesize fuel salt on-site as part of operations.
(b) Fresh fuel receipt (unirradiated): Since the fuel salt will arrive in a solid state, homogeneity would be a concern for NDA as well as sampling for destructive analysis.
(c) Transfer of fresh (unirradiated) fuel salt to a fuel storage area: At this point, the fuel salt would still likely be solid and containerized.
(d) Transfer of fresh fuel salt to a reactor primary loop: At this point, the fuel salt would be heated to its molten state and transferred to the fuel/coolant circuit.
(e) Fuel processing and polishing to remove volatile gasses and fission products: Different designs have different levels of fuel salt processing, and each system would be a potential take-off point for nuclear material. In addition, nuclear material will potentially be captured by filters and other cleanup equipment.
(f) Flush salt systems that will contain nuclear material as part of system clean out.
(g) Transfer of irradiated fuel salt to a salt storage area and shipment of irradiated fuel: The salt will exit the reactor systems as liquid salt and will then be placed in storage containers, where it would presumably solidify. Homogeneity would be a concern for NDA and sampling for destructive analysis.
(h) Other waste streams containing nuclear material: There may be other waste streams associated with operations, including fuel salt leakage and cleanup of the salt, as well as equipment that is redundant and contains nuclear material.

On the basis of the flow of nuclear material in a liquid fuelled MSR, some of the key issues with measurement points and applying IAEA safeguards include the following:

(a) The potential variability of the continuous or batch addition of fissionable, fertile or fissile material. This will change the reactor dynamics and make power monitoring more challenging.

(b) On-line processing and continuous or batch removal of material. These represent challenges for multiple points where nuclear material could exit the reactor system.
(c) Establishing material balances and conducting physical inventories for an operating bulk facility that is rarely shut down.
(d) Maintaining continuity of knowledge for bulk materials as they are transferred from the fresh fuel area to the fuel or coolant circuit, to spent fuel and waste streams.
(e) Developing new modelling tools to predict the inventories of nuclear material.
(f) Changing conditions during the lifetime of an MSR. There may be significant changes in operating parameters in progressing from a startup to an equilibrium core. Moreover, some MSRs are intended to be designed to change their operating parameters during the life of the facility — for example, changing the input of nuclear material to go from a breeder to a burner. This kind of flexibility would pose safeguards challenges because the operations could introduce changes in the nuclear material inventory. Thus, the safeguards verification activities may need to change significantly to address the variations of these reactor operating parameters.
(g) Lack of access by IAEA inspectors owing to extreme physical and radiation environments. High MSR operating temperatures (typically 600°C and more) and high radiation levels (delayed neutrons and activation products) in sections of the fuel/coolant circuit outside the core will limit access to reactor areas.
(h) Developing the appropriate NDA and destructive analysis techniques and measurement uncertainties.
(i) Holdup of nuclear material in systems and equipment. Operating experience from the United States programme at the Oak Ridge National Laboratory in the 1950s and 1960s indicates that remote manipulation and maintenance will be necessary, and components or entire sections of the reactor system will be designed to be removed and replaced remotely.
(j) Homogeneity. If the liquid salt is being sampled for destructive analysis, its homogeneity would have to be considered. In addition, if the salt is not kept heated, it will solidify. Once solidified, fuel homogeneity would likely become an even greater issue if NDA or sampling for destructive analysis is desired.
(k) Fuel salt chemistry and isotopic effects on fissile material production. Another unique feature of liquid fuelled MSRs is that the fuel salt chemistry may affect the reactor neutronics and, therefore, the rate of production and fission of fissile isotopes. For example, certain chemical properties of the fuel salt may result in different actinide containing compounds to plate

out inside the primary loop. In addition, fuel salt chemistry will affect the solubility of fission products and their rate of buildup, which affects parasitic losses and this in turn changes the operating parameters of the system.

5.1.8.3. Identified areas for additional consideration

The following areas for additional consideration were identified:

(a) The IAEA has experience in monitoring the output of reactors with solid fuel that is fixed, but for designs such as MSRs and PBRs, the reactor operating modalities may be significantly different from the current fleet of WCRs and even PHWRs, which are on-line fuelled reactors covered in IAEA safeguards. One of the main differences is that reactivity in solid fuelled WCRs is limited by fuel burnup so periodic refuelling is necessary. Reactors with moving fuel may never need to be shut down for refuelling, since one of their principal design features is continuous on-line refuelling.
(b) Since measurements and material accountancy will be more challenging compared with WCRs, the use of containment and surveillance will likely be of greater importance. However, a liquid fuelled MSR is not a high throughput facility, so there is not a large amount of nuclear material entering and exiting the facility. As such, the inventory can be relatively tightly controlled, since there will be much smaller 'processing' losses (chiefly via off-gas and cleanup systems). Consequently, measurement uncertainties will be greatly reduced compared with other high throughput facilities (e.g. reprocessing or fuel fabrication facilities). This factor is important for the measurement of the nuclear material, as it reduces the challenges of closing material balances, given that the volume of nuclear material entering and exiting the facility during normal operations is significantly smaller.
(c) Other areas that will likely become more important is the IAEA's use of facility operated equipment and instruments, such as for monitoring the operational status of the reactor, process monitoring and in-core monitoring. Additionally, extended use of unattended and remote monitoring will likely be needed because of the limited access and the high radiation and high temperature environments present. Moreover, it needs to be determined how closely coupled the reactor operations are with the isotopic inventory so that the operational history can be used to indicate the content of special fissionable material.

All of these topics are to be considered for an SBD approach as part of an integrated approach that includes security, safety and other design considerations.

This topic is also cross-cutting with other topics such as new fuels, limited access and non-power applications. As already mentioned, EIDs may use fuel that differs from that used by current power reactors covered by IAEA safeguards; they may also involve high radiation fields, which limit access to the moving fuel, and high temperatures compared with conventional reactors that allow high heat applications.

5.1.9. Lack of measurement technologies and other verification techniques

Variable fuel cycles and operating parameters, inaccessibility by inspectors, high temperatures and radiation fields, remote locations, intermittent operations and moving fuel may require the application of new measurement techniques for safeguards purposes.

5.1.9.1. Identified areas for additional consideration

The ability to perform or verify the operator's measurements of nuclear material, to draw material balances and to determine unaccounted material is central to the IAEA's safeguards measures, and the lack of applicable measurement techniques requires further consideration. This is a cross-cutting issue that applies to many other topics (such as the lack of established measurement techniques for MSR fuel and for PBR fuel burnup measurements).

5.2. SECURITY CONSIDERATIONS

This section presents an assessment of the impact of EID areas of novelty on existing IAEA security approaches, as described in Section 2.4. The section is organized by topical area. For each topic, areas of novelty are explained from a security point of view. Then, potential implications for IAEA security approaches and key components of physical security — including the ability to detect, delay and respond — are outlined.

5.2.1. Design basis threat and physical protection systems

The characteristics and size of the facility are important considerations for determining and prioritizing the required security measures and PPS. These characteristics include the type, amount and physical state of the nuclear material present and the identification of vital areas and critical targets from the point of view of security. A security risk assessment needs then to be conducted to provide understanding of the security measures needed to protect the facility against the design basis threat prescribed by the State. The information in the DBT is considered to be classified and is shared with operators but not usually with designers or vendors, although a redacted version may sometimes be published. Nevertheless, a regulatory body may wish to share DBT information with the designer so that special security features may be incorporated into the design. In this case, the designer will probably need to be security vetted, and arrangements will need to be made to ensure that information is shared only on a 'need to know' basis. The need for sharing information early in the process may be more critical if the same EID design is planned to be sold to several countries.

The PPS for a nuclear facility such as an EID is specified in the design, and it needs to provide sufficient detection, delay and response in a coordinated way so as to nullify the DBT.

5.2.1.1. Identified areas for additional consideration

The following areas for additional consideration were identified:

(a) Regardless of any difficulties in gaining early access to the DBT, the physical security regulations for existing NPPs will apply to EIDs.
(b) Strategies to maintain security compliance, despite the general trend to reduce operating and staffing costs, are the main challenges for the application of security practices to EIDs [107, 111, 112]. This area is not covered by IAEA publications and, hence, may need additional consideration.

5.2.2. Transportability

All transport security measures, mentioned in the existing transport security series IAEA publications are applicable to EIDs [110–113, 134–136], although there are some areas for additional consideration.

5.2.2.1. Identified areas for additional consideration

The following areas for additional consideration were identified:

(a) Generally, transport security needs to consider multiple issues, including uncertainties arising from transportation through the public domain, involvement of multiple national, regional, international agencies with multiple security interfaces and the possibility of attack anywhere along the route (which may extend to thousands of kilometres).
(b) All security measures for nuclear facilities [107, 111, 112] and nuclear material in transport [134, 135] are applicable to TNPPs but some associated

security challenges may need additional consideration. In the scenarios where a TNPP is marine based (e.g. docked at a port), elements of both nuclear security and maritime security are to be applied. Following the evaluation of the threat conditions to TNPPs and the associated facilities, the implementation of physical protection measures is based on the graded approach for detection, delay and response to any attempts of malicious actions against the TNPP and the associated facilities. These physical protection measures may also include the designation of security areas, which combine the areas designated for nuclear security with the maritime security requirements. Implementation of physical barriers might be given priority [112, 137] (e.g. engineered floating barriers against intrusion of unauthorized water-borne crafts). For these scenarios, maritime and nuclear security are interdependent, emphasizing the need for good communication between relevant groups of professionals.

5.2.3. Remote locations

Some EIDs are expected to be deployed to remote locations.

5.2.3.1. Identified areas for additional consideration

The following areas for additional consideration were identified:

(a) Remote sites can present challenges to nuclear security because in the event of an attack by an adversary, it may be difficult for any off-site response force to access the site in a timely manner. The number and capabilities of armed response forces on the site may vary from one site to another according to the complexity of the site and the site's security measures to detect and delay adversaries. Provision of an on-site response force for every site may cause additional operational cost.
(b) There may also be an increased vulnerability during the transport of nuclear material or transport of a fully fuelled TNPP to the remote location.

5.2.4. Non-electrical applications, location in urban areas

Some EIDs are intended to be used for non-electrical applications, especially for industrial or district heating. The latter may require these EIDs to be placed close to urban areas for better utilization of the heat generated. This may challenge the physical protection arrangements around the EID and make it difficult to protect them. The size and shape of the protection zones around the reactor will be most affected.

However, this may be compensated by using additional security measures, for example, electrical fences, roaming patrols by guards accompanied by sniffing dogs at scheduled and unscheduled hours, hardened guard posts around the peripheral boundary, search lighting arrangements, hardened tactical locations inside the protected area and security traps for both personnel and vehicles.

5.2.4.1. Identified areas for additional consideration

No areas for additional consideration were identified.

5.2.5. Compact layout of EIDs and underground location

The underground construction of some EIDs may enhance their protection against certain events, such as aircraft attacks and sabotage. However, the underground location may also result in vulnerabilities to some external hazards that need consideration in security arrangements. Moreover, special arrangements may be needed so that emergency crews could have access to underground reactors in the event of a serious accident. In this context, security can be further enhanced by reducing the number of access paths to the reactor's internal areas to reduce the accessibility and number of potential sabotage targets [107, 111, 137]. Compact designs might, however, reduce the space available to implement redundant and independent barriers from a security perspective [107, 111, 137].

5.2.5.1. Identified areas for additional consideration

From a security perspective, the evaluation of the impact of a compact layout and underground construction is an area for additional consideration.

5.2.6. Protection requirements for new types of fuel and new fuel cycles

The choice of nuclear fuel design in EIDs will affect the security aspects (theft and sabotage) via such factors as: (a) the level of enrichment, which will affect the categorization of the nuclear material; (b) the characteristics of any nuclear or other radiological material that could affect an off-site release following a sabotage event; (c) the ability of a release to spread (e.g. whether it can be aerosolized); (d) its attractiveness for theft in terms of practicalities (amounts, radiation, physical form, dilution, solids/liquids, frozen); (e) the frequency of refuelling; and (f) the management of irradiated fertile materials in fast spectrum EIDs.

Certain EIDs, such as SMRs, have an amount of radiological material available for potential release that, in the case of a sabotage, would be much lower than that of a conventional WCR. These EIDs could have physically smaller cores or use TRISO fuel, which is hard to aerosolize. However, the modularity of EIDs is to be taken into consideration when implementing security measures to prevent sabotage, as more than one SMR in the same site can have an amount of radiological material that is similar, if not higher, than for a conventional WCR, and modularization increases the number of targets.

5.2.6.1. Identified areas for additional consideration

All IAEA Nuclear Security Series publications are applicable to these new fuels and the corresponding fuel cycle. Nevertheless, as the development of new fuel types and fuel cycles progresses and specific handling arrangements are developed, additional security considerations may emerge from more detailed analysis.

5.2.7. Graded approach based on potential consequences

The main issue for all NPPs is to develop security requirements that consider radiological consequences and health impacts to the public in case of a radiological release and to establish physical protection levels following a graded approach. When applying the graded approach, certain EIDs might have less nuclear material and smaller vital areas than conventional NPPs. Hence, they may be considered less attractive installations for a perpetrator. Nevertheless, the operators of NPPs, including EIDs, still need to maintain the national requirements for physical protection to deter, detect, prevent and respond to any attempts of theft or sabotage.

The use of a graded approach facilitates the application of proportionate physical protection measures, taking into account the current evaluation of the threat, the relative attractiveness, the nature of the nuclear material and the potential consequences of the unauthorized removal of nuclear material (determined using the categorization of nuclear material provided in table 1 of Ref. [107]) and of sabotage (determined using an approach of grading radiological consequences). Certain EID designs might have lower fissile inventories inside the core compared with a conventional NPP. The amount of spent fuel will also depend upon how frequently the EID is fuelled and whether it has fuel in its core for its lifetime [107, 111, 112].

5.2.7.1. Identified areas for additional consideration

The graded approach is covered by IAEA Nuclear Security Series publications (e.g. Ref. [107]) and it is an important principle for EIDs. The practical implementation of the graded approach might be an area for additional consideration.

5.2.8. Insider threats

Nuclear security measures implemented on NPPs, including EIDs, need to be effective against threats by an insider. IAEA Nuclear Security Series No. 27-G, Physical Protection of Nuclear Material and Nuclear Facilities (Implementation of INFCIRC/225/Revision 5) [111], states:

> "An insider is defined as one or more individuals with authorized access to nuclear facilities or related sensitive information who could attempt unauthorized removal or sabotage or who could aid an external adversary to do so. An insider threat is an insider with an intention to carry out such an act. Insiders may include managers, regular employees, contractors and service providers, inspectors and some visitors. An insider may therefore be in any position at a facility and may have authorized access to any of the controlled areas or materials."

The attributes of an insider are typically defined by the extent of their authorized access, level of authority and knowledge. It is also possible that an individual with authorized access is unknowingly providing information to the perpetrators [111]. This can be described as an unwitting insider, that is, "an insider without the intent and motivation to commit a malicious act who is exploited by an adversary without the unwitting insider's awareness" [112].

Insider threats present different problems from external adversaries because they can take advantage of the information, access and/or authority privileges to bypass some technical and administrative physical protection measures to commit or facilitate unauthorized removal or sabotage. Along with understanding the attributes, insiders may have one or multiple motivations to initiate a malicious act. Insiders can also contribute to a malicious act through a series of separate actions (protracted theft[34] or sabotage) over an extended period of time, which may reduce the chance of detection and therefore increase the likelihood of success. Abrupt theft of nuclear material is when the amount of unauthorized material removed is significant and during a single event. Insiders may also have more knowledge and/or opportunity to select the most vulnerable target and the best time to perform the malicious act and may be in a position to increase the impact of an external attack [111, 112].

---

[34] 'Protracted theft' is the repeated unauthorized removal of potentially small quantities of nuclear material from either a single location or multiple locations." [112]

Preventive measures (e.g. identity verification, trustworthiness assessment, escort of infrequent workers, security awareness) are aimed at precluding possible insider threats, minimizing threat opportunities and preventing malicious acts from being carried out. Protective measures (e.g. security sensors, monitoring of personnel, two person rule, tracking movement, physical barriers, event investigation, emergency planning) aim to detect, delay and respond to malicious acts that are carried out and to mitigate or minimize their consequences. The preventive and protective measures against insider threats are supported by stringent nuclear material accounting and control measures [111, 112]. The nature of insider threats is highly dependent on the number of employees. The number of operational staff working at an EID will often be smaller than at a conventional WCR, and this may increase insider risks because the operator will rely on fewer individuals with more roles, responsibilities and authority. These individuals could become ‘super insiders’. When the human element is the weakest link in the security chain, it becomes a more attractive source of information for adversaries. Understanding and monitoring the insider threat challenge will therefore remain an important element of the physical and cybersecurity programme(s) of the EID.

IAEA Nuclear Security Series publications and recommendations provided in the IAEA publications in this area are applicable to EIDs.

5.2.8.1. Identified areas for additional consideration

No areas for additional consideration were identified.

5.2.9. Cybersecurity considerations for EIDs

Cybersecurity is vital for EIDs to protect their operations from both the security and safety points of view, as digital based systems are very widely applied in the security and safety systems of these reactors. A cyberattack could have an immediate impact, causing damage to equipment or degradation in security functions, such as covert information collection. It could also include a delay, producing a timed or separately triggered effect, and could be synchronized with other adversary activities, which may include a physical attack. Member States are responsible for providing the requirements on cybersecurity and for ensuring that operators provide adequate protection of digital based systems against cyberattacks. In this context, the overall goal of cybersecurity is to protect digital based systems against attacks aimed at facilitating the unauthorized removal of nuclear material or sabotage of nuclear facilities.

The EID designs imply extensive use of digital I&C, which creates additional challenges for the cyber security of these reactors [138, 139].

IAEA Nuclear Security Series publications in this area are applicable to EIDs.

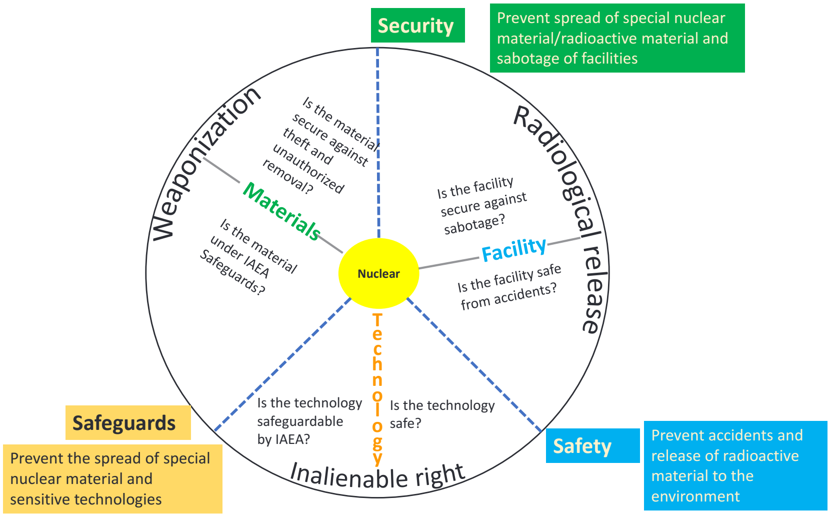


FIG. 2. Visual representation of interfaces between nuclear safety, security and safeguards.

5.2.9.1. Identified areas for additional consideration

No areas for additional consideration were identified.

5.3. SAFETY, SECURITY AND SAFEGUARDS INTERFACES: POTENTIAL SYNERGIES AND CHALLENGES

5.3.1. General

Applying an integrated approach to safety, security and safeguards is especially important in the conceptual design stage, when there is an opportunity to consider all three elements in an effective and optimized manner. For this, it is important to thoroughly understand and manage interfaces between safety, security and safeguards in order to mitigate conflicts, if any, and try to achieve synergies whenever possible.

The concept of 3S interfaces is visualized in Fig. 2. When 3S is considered holistically, as illustrated in Fig. 2, the interfaces between each of the 2S pairs are related to: facility (safety–security), materials (security–safeguards) and technology (safeguards–safety). These are the potential domains where the synergy between safety, security and safeguards needs to be considered. For instance, both nuclear safety and security focus on the facility (e.g. EID) with the objectives of protecting people and the environment from potential radiation exposure. In the domain between security and safeguards, the nuclear security regime prevents adversaries, including terrorists and insiders, from theft and unauthorized removal of such materials, which supports the purposes of the safeguards and prevents nuclear weaponization. Finally, the technology is the focal point between safeguards and safety, and it must be ensured that it is safe and can be safeguarded without compromising one of these elements. The interfaces depicted in Fig. 2 lay the groundwork for a holistic approach to integrate 3S in the early design stages of EIDs.

This section addresses each of the 3S interfaces:

(a) Safety–security;
(b) Security–safeguards;
(c) Safeguards–safety.

For each interface, general areas where synergies might be found are examined. This is followed by a description of potential challenges and possible issues related to EIDs that may need further consideration. This section does not present solutions that would capitalize on the synergies or resolve the challenges; rather, it highlights the aspects that need to be examined for that purpose. Finally, Section 5.3.4.4 provides some general conclusions regarding the 3S concept for EIDs.

5.3.2. The safety–security interface

5.3.2.1. General synergies

Some of the areas for potential synergies between safety and security are briefly provided below:

(a) Management systems. Promoting integrated management systems at EIDs, through an effective legislative and regulatory framework, is necessary to address specific aspects dealing with both security and safety (e.g. policies, requirements, quality assurance, measures).
(b) Regulatory framework. There are numerous parallels and interactions between security and safety in the context of the legal and regulatory frameworks. Regulations in terms of safety and security measures that need to be implemented at NPPs, including EIDs, for controlling risks (regardless of whether they arise from safety or security events) are potential areas for synergies.
(c) Graded approach. Application of a graded approach for EIDs with consideration of the level of risk posed by these reactors is relevant for both

safety and security considerations. The factors used in the context of a graded approach are different for safety (reactor power level, fuel characteristic, inventory of radioactive material) and security (type of radioactive material, attractiveness), but they could have common considerations (e.g. radiological consequences), which could be the areas for potential synergies.

(d) DiD. The DiD principle is applicable to both safety and security and needs to be utilized for both safety events and security incidents by applying several layers of barriers and levels. The implementation of the DiD principle represents an area where synergy between safety and security could potentially be achieved.

(e) Emergency response. Plans for emergency preparedness and response dealing with nuclear emergencies, including nuclear emergencies triggered by a nuclear security event, need to be developed in a coordinated and integrated manner with respective security plans. This will allow for effective actions between various teams dealing with the safety and security aspects of the response and is an area for potential synergies.

(f) Use of safety analysis insights to support security considerations. Outcomes from safety analysis performed for EIDs provide an important input for informing and enhancing the security arrangements for a given reactor design. In particular, the information from safety analysis (e.g. risk insights) could be used for security considerations such as vital area identification, target sets definition or potential security event consequence analysis. This will allow optimizing the security arrangements while designing EIDs and apply a graded approach where applicable.

5.3.2.2. General challenges

The general challenges are as follows:

(a) Prioritization in case of conflicting interfaces. The Safety Fundamentals stress the need for safety and security to work cooperatively to protect human health and the environment. Some of the interfaces between safety and security might be conflicting, where concrete decision making might require a trade-off between safety and security considerations. Such kinds of decision are challenging and there is no clear guidance on how to prioritize safety and security considerations.

(b) Access control. Most nuclear facilities operate on a continuous basis and access/egress is controlled according to standard security procedures. There may be times, however, when it will be necessary to suspend these procedures so as to allow, for example, repairs or inspections to take place by outside parties. One would expect the security procedures to anticipate such circumstances, but it is difficult to cover every eventuality and there may be pressure — for safety or other reasons — to circumvent the procedures so as to complete the work quickly.

(c) Emergencies. The occurrence of a nuclear emergency might lead to the evacuation of (non-essential) on-site personnel. From the security point of view, however, such protective action might constitute a potential vulnerability (e.g. potential risk for nuclear material to be smuggled out of the facility). Such a situation could be exploited by an insider adversary who could set off an alarm for this purpose.

(d) Transparency. Because potential safety issues need to be widely understood by designers and operators, it is expected that safety will be an open and transparent practice. In contrast, security usually leans towards confidentiality and control of information. This can lead to conflicts, as when a decision is needed on how widely to reveal the purpose of individual buildings in a large site.

5.3.2.3. Identified areas for additional consideration

The following areas for additional consideration were identified:

(a) For a floating TNPP, it is challenging to define the interface between safety and security, since the floating TNPP is a nuclear facility as well as a maritime vessel. As a nuclear facility, it might use other design features that are not common to conventional WCRs. As a maritime vessel, the floating TNPP is subjected to weather disturbances such as typhoons, cyclones or hurricanes, tsunami, plus sea-borne security threats. It would also have to comply with laws and regulations depicted by the IMO.

(b) EIDs with highly integrated software based systems have specific benefits and challenges in terms of safety and security. The architecture of these systems allows for better flexibility, offers wide computational capabilities, fast processing of complex data, enhanced diagnostic capabilities and other benefits from a safety perspective. At the same time, such systems might increase the threat from cyberattacks. For instance, the target for an attack could be the fuel loading systems, especially for EIDs with continuous fuelling.

5.3.3. The security–safeguards interface

5.3.3.1. General synergies

Some of the areas for potential synergies between safeguards and security are briefly described below:

(a) Access control systems. Access control to EIDs in all plant states might be a potential area for synergies between security (e.g. only authorized personnel) and safeguards (e.g. access of safeguards inspectors). In this context, the synergies might be achieved through administrative or technical measures to monitor the access of individuals within the EID and control access to the nuclear material itself.

(b) State system of accounting for and control of nuclear material (SSAC). For nuclear security, the goal of the SSAC is to account for and protect material from unauthorized removal and sabotage, as well as resolve inventory discrepancies and respond to unexplained losses. A facility may be divided into several MBAs to provide better control of nuclear material for security purposes[35]. For IAEA safeguards, the goal is to verify that the State has not diverted and/or misused nuclear material and technology. For IAEA safeguards purposes, an MBA is an area inside or outside a facility such that the quantity of nuclear material transferred into or out of the MBA can be determined. Therefore, the goals and loss detection approaches are similar for both security and safeguards programmes and is a potential area for synergies.

(c) Timely detection. For security purposes at a facility level, a domestic SSAC system aims to detect, as fast as possible, the removal of smaller quantities of material and provide assurances that nuclear materials are being used for their intended purposes. IAEA safeguards rely on a timely detection when a significant quantity of material is diverted. Thus, the timely detection measures might be an area of synergy between security and safeguards.

(d) Inspections. The inspection objectives for security and for safeguards purposes are different. Security inspection verifies the ability of the facility's nuclear security system to implement the nuclear material measures, prevent theft or sabotage, detect in a timely manner theft or unauthorized removal through verification of compliance of State rules and license conditions; whereas the objective of a safeguards inspection is to verify non-diversion of nuclear material through verification measures. While the objectives of

[35] Not all States use the MBA terminology for nuclear security.

these inspections might differ, the findings and lessons learned from each of these inspections might enhance the other area.

5.3.3.2. General challenges

Interfaces between security and safeguards are generally mutually supportive, as evidenced by the fact that some regulatory bodies choose to combine the two. No challenges of a general nature were identified.

5.3.3.3. Areas for additional consideration in relation to EIDs

Some EIDs with an increased level of fuel enrichment would increase the attractiveness of unauthorized removal of their nuclear material for adversaries and, hence, would be of specific security and safeguards concern. For EIDs of the TNPP type, the spent fuel would become a focal point of security and safeguards. Protecting spent fuel stored on board would be challenging for the State's domestic security programme. Although the physical protection of a floating TNPP is confined by its hull, the physical protection perimeter is around its operating site. In comparison with land based NPPs, a floating TNPP has a part of water area around itself that needs to be defended, and prevention of a sea-borne attack by adversaries requires further scrutiny. In this context, such an attack might be challenging to tackle.

On the safeguards side, it would be more cumbersome for IAEA safeguards inspectors to perform inspection and surveillance of spent fuel stored in a ship (in the case of potential concepts with on-vessel spent fuel storage). However, if the spent fuel can be returned to the vendor State (e.g. nuclear weapon State), the burden of securing the spent fuel by the recipient and safeguarding them by the IAEA would be lessened.

A number of EIDs (e.g. pebble bed concepts, MSRs, factory-sealed cores) might require new technologies to be developed to support security and safeguards provisions in the plant systems. In particular, for on-line refuelling, a safeguards approach would have to be developed to monitor the movement of the fuel. This approach ensures that there is no diversion or misuse of the nuclear material refuelling.

Some EID concepts are intended to be deployed in remote locations with limited or no human presence on-site. In this context, the safeguards measures and monitoring of the safety parameters might require the continuous transfer of surveillance and operational data from the facility, which is considered to be a potential vulnerability from security perspectives.

5.3.4. The safeguards–safety interface

5.3.4.1. General synergies

Some of the areas for potential synergies between safeguards and safety are briefly discussed below:

(a) Legal frameworks. For some EIDs, such as micro-size reactors, there are different deployment models, including different roles between design owner, operator, transport organizations and other interested parties. This situation may present different complexities and challenges towards safeguardability by the IAEA in accordance with SBD. Therefore, the synergies might be achieved by universalization of existing international and national legal and institutional frameworks. This might allow the identification of possible innovative approaches and other efforts needed to address such nuclear exports.
(b) Design. The design process, which follows the design safety requirements, needs to consider also the interfaces with safeguards (see Requirement 8 of SSR-2/1 (Rev. 1) [3]). During the design stage of an EID, when fulfilling the design safety requirements, arrangements could be made to facilitate the implementation of international safeguards (e.g. accessibility to nuclear material, providing reliable and optimized support to surveillance systems). Thus, consideration of safeguards while implementing the design safety requirements is an area of synergy between safety and safeguards.
(c) Fuel manufacturing. The fuel manufacturing process to assure reliability of the fuel is a key aspect for the safe operation of EIDs, which is also an important consideration from a safeguards perspective. For instance, the amount of failed fuel elements needs to be kept to a minimum during pebble bed HTGR operation. As a result, the IAEA safeguards effort during reactor operation can be lessened if the amount of failed fuel elements is reduced. Thus, fuel manufacturing is an area of synergy between safety and safeguards.

5.3.4.2. General challenges

The general challenges at the safeguards–security interface will be broadly similar to those found at the safety–security interface, namely: (a) access control; (b) emergencies; and (c) transparency (see details in Section 5.3.2.2, item (d)).

In addition, the following general challenges are related to the interface between safeguards and safety:

(a) As stated in para. 87 of Ref. [108], it is important that the implementation of safeguards inspection activities does not affect the safety of the facility;
(b) The access of safeguards inspections in specific cases might be delayed or prevented because of safety considerations.

Consideration of these general interfaces in the early design stages will allow addressing these challenges in an effective manner through dedicated solutions incorporated into the design or through organizational and administrative arrangements.

5.3.4.3. Areas for additional consideration in relation to EIDs

Some aspects at the safeguards–safety interface require early expert understanding of the EID designs that could be developed. Examples include the following:

(a) For EIDs, an early integration of safeguards issues into the design is desirable but will require international cooperation between the IAEA and all EID stakeholders (e.g. State authorities, regulatory bodies, designers, equipment providers). This presents a considerable challenge.
(b) Reactors operating with a compact configuration of structures, systems and components may create conflicts between safeguards activities and safety procedures. Hence, specific provisions are needed to ensure that the safety of workers and inspectors is not compromised during safeguards inspection and that safety measures would not hinder the IAEA safeguards personnel in conducting an independent and impartial inspection. In particular, the configurations for TNPPs may pose specific challenges because of the need to incorporate transportability requirements into the design (e.g. the TNPP needs to fit on a truck or train, to be able to pass through tunnels or over bridges, or to be adapted into a vessel that is sufficiently stable on water). Examples of provisions include clear markings of access and passages, coordinated safety escort, well defined MBA and key measurement points, and precise physical inventory with continuity of knowledge for verification.
(c) For all reactors, the amount of failed fuel in operation needs to be kept to a minimum. In the case of PBRs, identifying and retrieving failed pebbles during operation is difficult. When a failed pebble exits from the core vessel, it may be sent to a collection bin for post-irradiation examination to determine the causes of failure. Difficulties in retrieving failed fuel

elements are not limited to PBRs; however, considering the large number of pebbles, new approaches and more resources might be needed to deal with this type of fuel from a safety and safeguards point of view. For instance, the accounting process might be complex, which will require additional efforts from the IAEA safeguards for inspection and verification.

5.3.4.4. General conclusions regarding the 3S concept for EIDs

On the basis of the discussions provided earlier in this section, the following general insights were derived in terms of the development and application of the 3S concept:

(a) The processes and considerations on integrating and achieving synergies between safety, security and safeguards in the design of EIDs are not covered in IAEA publications.
(b) Incorporating the 3S concept during the design stage of an EID might require overcoming additional challenges of meeting national and international requirements at different stages during the design and construction of the EID. However, the preceding discussion highlights the benefits of integrating safety, security and safeguards and, if successfully implemented, such integration will allow all interested parties to optimize their workflows during the entire lifetime of EIDs.
(c) Therefore, the 3S concept needs to be initiated early and needs to continue throughout the design stage. With many of the EIDs still on the design stage, promoting and ensuring this objective is of interest to all interested parties.

6. KEY OUTCOMES OF THE APPLICABILITY REVIEW OF SAFETY STANDARDS

The review presented in this publication has confirmed that the safety standards are applicable to EIDs with some exceptions, as explained below. The existing safety framework described in the safety standards contains the tools to assess and regulate the safety of an EID, but some modifications or additions may be necessary to supplement the IAEA Safety Standards Series publications to address in an equitable way the full range of reactor technologies being considered. The tables in Appendix I present the identified needs to supplement the safety standards.

The review considered a wide range of EIDs, including water cooled SMRs, SFRs, LFRs and HTGRs. The review only partially considered MSRs, TNPPs and micro-size reactors (<10 MW(e)); for these technologies, there could be additional considerations not covered in this publication.

The review has identified important applicability considerations (including areas of non-applicability, gaps and other considerations) in some of the safety standards, particularly for the design and safety analysis for non-WCRs and the design and transport of TNPPs.

In several cases, careful consideration of the intent of the current text of the safety standards might be necessary for application to EIDs. There is also a lack of experience on the implementation of the existing safety framework to the different EIDs in line with the safety standards.

Areas of the safety standards that may not apply to EIDs tend to be very focused on large, land based, water cooled NPPs, where a technology neutral interpretation is difficult or not possible. In some cases, some of the requirements and recommendations could be modified, however, or new requirements and recommendations could be added to improve the applicability of the safety standards to a wide range of EID technologies; for example, by clarifying the intent of the requirements and recommendations in a technology neutral manner. The review did identify some instances where direct application of the safety standards (as currently written) to some EIDs, without consideration of the intent, could lead to incorrect results.

The other important applicability consideration is the existence in EIDs of areas of novelty that have safety implications but are not captured in the current safety standards. The future consideration of these areas of novelty may result in new and modified requirements and recommendations. A review of operating experience of past and current prototype and demonstration reactors would contribute to the development of relevant knowledge that could be used to update the safety standards. In some cases, there may be sufficient knowledge to revise the safety standards in the near and medium term. For other technologies, for which there is less knowledge and experience, a longer time frame will be needed for the state of knowledge to increase and consensus based safety standards to be developed. In some cases, technology specific recommendations may be necessary.

Key findings in the different technical areas are summarized below.

(a) Siting

The safety standards related to siting are applicable to EIDs. However, they do not cover TNPPs if these are not stationary at a site. The safety standards do not provide explicit guidance on the application of a graded approach for site evaluation, which may be of particular importance to SMRs and microreactors.

(b) Design and construction

The safety standards related to design and construction are applicable to EIDs, although careful consideration of the intent underlying the current text of the safety standards will be necessary. There are, however, gaps regarding several EIDs as explained below.

Some of the areas of the safety standards focus on conventional WCRs and therefore do not cover some of the design of EIDs. For example, EIDs can use novel fuels, new coolants and can have different confinement provisions. EIDs tend to rely extensively on passive systems and inherent safety features and modularity. EIDs can include multiple modules and related HSI design features. Some EIDs may be designed to enable alternative operating models such as factory construction, commissioning and refuelling, as well as remote operation.

In some cases, there is a lack of experience on how the safety standards apply to EIDs and therefore there is a lack of consensus on adequate interpretations for these designs. For example, on the concepts of 'severe accident' and 'design extension conditions' and the implementation of DiD level 4 and level 5, interpretations of these terms, design provisions and safety features can be different from those used for conventional WCRs. There is also a scarcity of experience on what PIEs are to be considered in the design and a lack of knowledge about the response of some EIDs to these PIEs.

An important finding of the review for all EID technologies is that there is a lack of regulatory and operating experience and a need to consider how this can be addressed, particularly when considering first of a kind technologies and designs.

EIDs may introduce phenomena, failure modes and hazards that are not relevant to WCRs and that are not considered in the current safety standards.

The standards also do not cover specific issues for the design of TNPPs related to the transport of the plant throughout its lifetime.

Furthermore, there are some gaps related to construction. For example, the safety standards do not consider novel or advanced manufacturing techniques, factory based manufacturing, common servicing and refuelling facilities or the potential increase in complexity in the supply chain.

(c) Commissioning and operation

The safety standards related to commissioning and operation are applicable to EIDs. Areas of novelty not covered by the safety standards include the use of alternative operating models, such as factory commissioning prior to transport to the operating site, autonomous operation, refuelling (possibly at the factory) and remote monitoring and intervention.

There are also implications resulting from novel design characteristics. For example, the need to verify the operability of passive safety systems, the implications of modularity, shared systems among modules and novel chemistry and hazards for commissioning and operation.

Particularly for first of a kind designs, setting and adapting OLCs for EIDs, considering the lack of experience and uncertainties, is addressed only generically in the safety standards.

(d) Leadership and management for safety

The safety standards in this area are applicable to EIDs. However, there are several issues that may merit additional work related to these technologies.

One of the crucial areas for leadership and management for safety is the deployment model, in particular, potential situations where the future owner, operator or both are not known in the phase of manufacturing. In such a situation, the future owner or operator cannot execute its oversight of the supply chain and is fully dependent on the oversight of the vendor. In practical terms, this means that the future owner or operator (i.e. the licensee responsible for nuclear safety) has no evidence to show how the responsibility for safety was implemented during manufacture.

There are several areas that could impact the management system, resources management, security aspects, etc. Potential examples are: (a) the staged construction and operation of modules, and (b) the consideration of alternative modes of operation, such as autonomous operation and remote monitoring and support centres.

(e) Safety of nuclear fuel cycle facilities

The safety standards in this area are applicable to EIDs and adequately address the hazards associated with new technologies. However, they do not cover some areas specific to novel fuel types and their designs.

Depending on a particular fuel type and its design, further technology specific guidance could be developed. The Safety Guides cover current fuel fabrication processes, but additional Safety Guides would need to be developed to cover the fabrication of advanced fuels and reprocessing of non-WCR fuel once there is sufficient knowledge and experience on these processes. Until such a point is reached, the guidance in SSG-43 [65] is seen as sufficiently general to guide the safety of nuclear fuel cycle facilities associated with EIDs.

(f) Safety assessment

The safety standards in this area are applicable to EIDs, however, SSG-4 [91] on Level 2 PSA is not fully applicable to non-WCRs. Some issues that may merit additional work include the definition of the term 'severe accident' for technologies for which core melting is claimed to be not relevant and how deterministic and probabilistic safety analyses are to be applied to EIDs.

The key general issue for safety analysis is related to the limited knowledge of physical phenomena, the scarcity of experimental and operational data, and the difficulty in addressing uncertainties. This is particularly relevant for first of a kind designs. Moreover, the interpretation of terms such as DBA, DEC or 'severe accident' will strongly influence the application of these standards, so the gaps for EIDs mentioned above are also relevant here. In addition, the safety standards do not explicitly include an approach with which to analyse the unique hazards associated with some of the EID technologies or to address important design differences (e.g. the consideration of functional containment concepts). Furthermore, some of the risk metrics typically used for WCRs, such as core damage frequency, may not be applicable to EIDs. For EIDs for which the consideration of core melt may not relevant, a specific safety guide on the probabilistic assessment of off-site consequences, which currently does not exist, might become important for the safety demonstration, particularly for DEC and severe accidents. Moreover, the safety architecture of some EIDs, such as HTGRs, mainly relies on a functional containment approach, inherently safe characteristics or extensive use of passive safety features. Such safety architectures can increase the need for an integral demonstration (with consideration of all the relevant safety systems and features) that their performance is adequate and relies less on qualifying and testing individually each safety system, as is usually the case for current WCRs.

(g) Radiation protection and safety

The safety standards in this area are applicable to EIDs but there may be specific challenges for different EIDs. Many of these are unknown at this stage and need to be further explored. There is a lack of experience on the implementation of the safety standards to different EID technologies. Examples include new source terms in operation and unique design features of EIDs, such as compact designs, the use of non-water coolants and different approaches to operation and other activities. For instance, maintenance or decommissioning might introduce challenges for the radiation protection of workers. In the case of public exposure, routine discharges are expected to be lower than those from existing reactors, but this may be dependent on the reactor design and operating conditions.

(h) Management of radioactive waste and spent fuel

At a high level, the safety standards in this area are sufficiently general to apply to all of the spent fuel and radioactive waste that could be generated from EIDs. In detail, however, there are a few areas that may benefit from additional consideration. For example, EIDs may lead to the generation of various novel spent fuel and waste types for which it may be appropriate to conduct further research and develop specific safety guidance. It may also be appropriate to develop or strengthen the safety guidance in several areas, including passive safety during radioactive waste management, processing of EID spent fuels for disposal, centralized storage of spent fuel, operational safety of waste and spent fuel management facilities at EID sites, and the disposal of entire reactors.

A general observation from the review is that so far, EID developers have devoted relatively little detailed attention to the management of radioactive waste and spent fuel from EIDs, and this highlights a general need for further research in this area.

(i) Decommissioning

At a high level, the IAEA safety standards in this area are sufficiently general to apply to decommissioning activities for many EIDs. However, the deployment of some EIDs will likely encourage the use of centralized facilities to undertake decommissioning tasks (as opposed to on-site decommissioning) and it is particularly relevant for TNPPs and EIDs with integrated reactor designs and one-time fuelling. The safety standards do consider the possibility of transferring the responsibility for decommissioning from the operating organization to another organization but only when this entails the handover of a site. The idea that major, disused items of the plant could be taken to another location for dismantling by a specialist entity is not considered. Furthermore, the current safety standards do not address the specific decommissioning tasks associated with EIDs when performed at a centralized facility.

(j) Emergency preparedness and response

The safety standards in this area are applicable to EIDs. However, there is a lack of experience in the application of the safety standards — especially those related to the assessment of hazards and potential consequences of an emergency, should it occur — to be considered when designing adequate on-site and off-site emergency arrangements for EIDs. This includes determining the EPC associated with the facilities operating EIDs, as well as the sizes and associated emergency arrangements of EPZs and EPDs. In this context, the definitions of EPC, EPZ and EPD for TNPPs during transportation require particular attention.

(k) Transport of radioactive material

The safety standards in this area are generally applicable to the transportation of EIDs but do not cover the case of the transport of EIDs containing fresh fuel, SNF, and contaminated and radioactive material.

(l) Legal and regulatory framework for safety

The safety standards in this area are applicable to EIDs. Important issues that may merit additional work are (a) the increasing importance of cooperation among regulatory bodies of Member States in view of limited resources and increased globalization, and (b) the necessary regulatory oversight of EIDs, considering the potentially different deployment models that might be used. For example, the arrangements for regulatory oversight of an EID that was fabricated in one country and operated in a different country. Regulatory bodies of Member States could benefit from sharing experiences, insights and coordinating actions related to all aspects of EIDs.

(m) Interface between safety, security and safeguards

The safety standards and security guidance in this area are applicable to EIDs. However, there is a lack of guidance regarding the application of the 3S concept for EIDs through the proper consideration of potential interfaces between safety, security and safeguards. The potential interfaces are indicated in Section 5 and described through examples. Thorough understanding of potential interfaces, conflicts and synergies between safety, security and safeguards is an important prerequisite for setting up the relevant measures (e.g. design solutions, organizational matters, regulatory framework) for effective integration of these three disciplines for EIDs and for avoiding minimizing potential negative interactions. Examples of areas of novelty that need further consideration relate to aspects specific to EIDs, such as transportability, new fuel concepts, limited access and remote locations, non-electrical applications and a potential increase of cyberattacks. With many EIDs at an early design stage, there is a unique time window to develop and to effectively utilize an integrated approach to safety, security and safeguards.

Appendix I

OVERVIEW OF SAFEGUARDS CHALLENGES

An overview of additional considerations and challenges related to the IAEA safeguards of EIDs discussed in Section 5.1 is shown in Fig. I.1. The table indicates the safeguards challenges relevant to various types of EID. The types of EID provided in the table are based on the technologies described in Section 1.4. These technologies are compared against the novelties associated to the safeguards challenges described in Section 5.1 (e.g. transportability, small size of fuel items, new fuels and fuel cycles, limited access, remote locations).

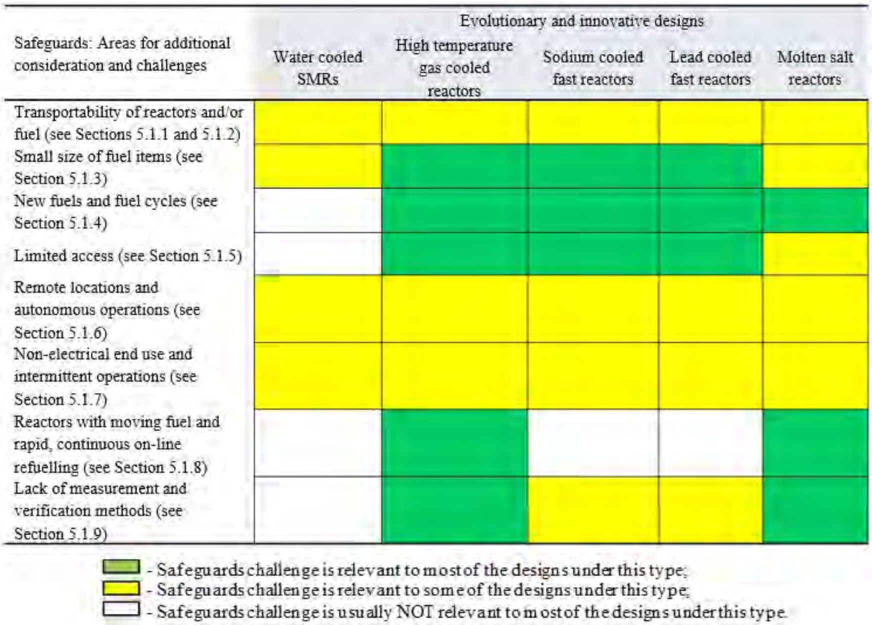
The following types of reactor have been considered in Fig. I.1:

(a) Water cooled SMRs;
(b) High temperature gas cooled reactors;
(c) Sodium cooled fast reactors;
(d) Lead cooled fast reactors;
(e) Molten salt reactors.

Transportable NPPs are not considered explicitly but as a part of different types of the above mentioned designs, which potentially could be transportable.

The information presented in Fig. I.1 is based on the discussion provided in Section 5.1 and is not intended to be exhaustive or comprehensive for all EID types under development, but only a high level representation of generic designs. Mapping of various designs and novelties is implemented in the following three categories:

— Safeguards challenge is relevant to most of the designs under this type;
— Safeguards challenge is relevant to some of the designs under this type;
— Safeguards challenge is usually **not** relevant to most of the designs under this type.



Safeguards: Areas for additional consideration and challenges	Evolutionary and innovative designs				
	Water cooled SMRs	High temperature gas cooled reactors	Sodium cooled fast reactors	Lead cooled fast reactors	Molten salt reactors
Transportability of reactors and/or fuel (see Sections 5.1.1 and 5.1.2)					
Small size of fuel items (see Section 5.1.3)					
New fuels and fuel cycles (see Section 5.1.4)					
Limited access (see Section 5.1.5)					
Remote locations and autonomous operations (see Section 5.1.6)					
Non-electrical end use and intermittent operations (see Section 5.1.7)					
Reactors with moving fuel and rapid, continuous on-line refuelling (see Section 5.1.8)					
Lack of measurement and verification methods (see Section 5.1.9)					

- Safeguards challenge is relevant to most of the designs under this type;
- Safeguards challenge is relevant to some of the designs under this type;
- Safeguards challenge is usually NOT relevant to most of the designs under this type.

FIG. I.1 Overview of safeguards challenges for EIDS.

—

Appendix II

SUMMARY TABLES

Figures II.1–II.12 provide a representation of the perceived needs to supplement the IAEA safety standards through an expert based evaluation of the extent of the considerations (including areas of non-applicability, gaps, other considerations) identified by the review in terms of the number of applicability considerations or impact on requirements or recommendations.

The review considered a wide range of EIDs, including water cooled SMRs, SFRs, LFRs and HTGRs. The review only partially considered MSRs, TNPPs and micro-size reactors (<10 MW(e)), and for these technologies there could be additional considerations which are not covered in this publication.

The information provided in these figures does not account for the 'severity' of the considerations identified. In some cases, even if a small number of considerations have been identified, addressing them may require significant future work. The extent of this work is not accounted for in the tables. Furthermore, there is a variability on the scope of the considerations depending on the technical areas; whenever possible, considerations were grouped in overarching findings.

The right column of each of Figs II.1–II.12 also provides a preliminary indication of whether the safety standards may have some limitations when applied to specific EID technologies; in some cases, these can be overcome by interpretation of the intent of the requirements and recommendations.

IAEA Safety Standards Series publication		Main limitation
SSR-1	Site Evaluation for Nuclear Installations	TNPPs not placed at a site
SSG-35	Site Survey and Site Selection for Nuclear Power Plants	
SSG-79	External Human Induced Events in Site Evaluation for Nuclear Power Plants	
NS-G-3.2	Dispersion of Radioactive Material and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants	
NS-G-3.6	Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants	
SSG-9 (Rev. 1)	Seismic Hazards in Site Evaluation for Nuclear Installations	
SSG-18	Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations	
SSG-21	Volcanic Hazards in Site Evaluation for Nuclear Installations	

Small number of applicability considerations or very small impact on the safety standard

FIG. II.1. Safety standards for siting.

IAEA Safety Standards Series publication		Main limitation
SSR-2/1 (Rev. 1)	Safety of Nuclear Power Plants: Design	Non-WCRs, TNPPs
SSG-30	Safety Classification of Structures, Systems and Components in Nuclear Power Plants	No limitations
SSG-52	Design of the Reactor Core for Nuclear Power Plants	Non-WCRs
SSG-56	Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants	Non-WCRs
SSG-53	Design of the Reactor Containment and Associated Systems for Nuclear Power Plants	Non-WCRs, TNPPs
SSG-34	Design of Electrical Power Systems for Nuclear Power Plants	Passive systems, multiple modules
SSG-39	Design of Instrumentation and Control Systems for Nuclear Power Plants	EIDs special features
SSG-51	Human Factors Engineering in the Design of Nuclear Power Plants	Multiple modules
SSG-62	Design of Auxiliary Systems and Supporting Systems for Nuclear Power Plants	Non-WCRs, passive systems
SSG-63	Design of Fuel Handling and Storage Systems for Nuclear Power Plants	Non-WCRs, TNPPs
SSG-64	Protection Against Internal Hazards in the Design of Nuclear Power Plants	Non-WCRs
SSG-68	Design of Nuclear Installations Against External Events Excluding Earthquakes	TNPPs not placed at a site
SSG-67	Seismic Design for Nuclear Installations	
SSG-69	Equipment Qualification for Nuclear Installations	Non-WCRs, passive systems
NS-G-1.13	Radiation Protection Aspects of Design for Nuclear Power Plants	Non-WCRs
SSG-38	Construction for Nuclear Installations	Factory build

No applicability considerations
Small number of applicability considerations or very small impact on the safety standard
Some applicability considerations or small impact on the safety standard
Numerous applicability considerations or more than a third of the safety standard impacted

FIG. II.2. Safety standards for design and construction.

IAEA Safety Standards Series publication		Main limitation
SSR-2/2 (Rev. 1)	Safety of Nuclear Power Plants: Commissioning and Operation	Non-WCRs, remote operation, multiple modules
SSG-28	Commissioning for Nuclear Power Plants	Multiple modules, shared
SSG-76	Conduct of Operations at Nuclear Power Plants	systems,
SSG-72	The Operating Organization for Nuclear Power Plants	factory construction
SSG-70	Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants	First of a kind designs
SSG-71	Modifications to Nuclear Power Plants	Multiple modules
SSG-73	Core Management and Fuel Handling for Nuclear Power Plants	Non-WCRs
SSG-74	Maintenance, Testing, Surveillance and Inspection in Nuclear Power Plants	Multiple modules, shared systems, remote operation
SSG-75	Recruitment, Qualification and Training of Personnel for Nuclear Power Plants	Multiple modules, remote operation, factory refuelling
SSG-13	Chemistry Programme for Water Cooled Nuclear Power Plants	Non-WCRs
SSG-48	Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants	Non-WCRs
SSG-50	Operating Experience Feedback for Nuclear Installations	No limitations
SSG-77	Protection Against Internal and External Hazards in the Operation of Nuclear Power Plants	Non-WCRs, remote operation
SSG-54	Accident Management Programmes for Nuclear Power Plants	Non-WCRs

No applicability considerations
Small number of applicability considerations or very small impact on the safety standard
Some applicability considerations or small impact on the safety standard

FIG. II.3. Safety standards for commissioning and operation.

IAEA Safety Standards Series publication		Main limitation
SSR-4	Safety of Nuclear Fuel Cycle Facilities	Advanced fuels, fuel reprocessing
SSG-27	Criticality Safety in the Handling of Fissile Material	No limitations
SSG-5	Safety of Conversion Facilities and Uranium Enrichment Facilities	Advanced fuels, fuel reprocessing
SSG-6	Safety of Uranium Fuel Fabrication Facilities	
SSG-7	Safety of Uranium and Plutonium Mixed Oxide Fuel Fabrication Facilities	
SSG-42	Safety of Nuclear Fuel Reprocessing Facilities	
SSG-43	Safety of Nuclear Fuel Cycle Research and Development Facilities	No limitations

No applicability considerations
Small number of applicability considerations or very small impact on the safety standard

FIG. II.4. Safety standards for the safety of nuclear fuel cycle facilities.

IAEA Safety Standards Series publication		Main limitation
GSR Part 3	Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards	No limitations
GSG-7	Occupational Radiation Protection	None identified at this stage
GSG-8	Radiation Protection of the Public and the Environment	No limitations
GSG-9	Regulatory Control of Radioactive discharges to the Environment	Underground EIDs
GSG-10	Prospective Radiological Environmental Impact Assessment for Facilities and Activities	Underground EIDs
RS-G-1.7	Application of the Concepts of Exclusion, Exemption and Clearance	No limitations

No applicability considerations
Small number of applicability considerations or very small impact on the safety standard

FIG. II.5. Safety standards for radiation protection and safety.

	IAEA Safety Standards Series publication	Main limitation
GSR Part 5	Predisposal Management of Radioactive Waste	Novel wastes
GSG-1	Classification of Radioactive Waste	Passively safe storage
SSG-40	Predisposal Management of Radioactive Waste from Nuclear Power Plant and Research Reactors	Processing for disposal
SSG-41	Predisposal Management of Radioactive Waste from Nuclear Fuel Cycle Facilities	Novel wastes
SSG-45	Predisposal Management of Radioactive Waste from the Use of Radioactive Material in Medicine, Industry, Agriculture, Research and Education	No limitations
WS-G-6.1	Storage of Radioactive Waste	Centralized storage
GSG-3	The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste	External hazards, sequential construction
SSG-15 (Rev. 1)	Storage of Spent Nuclear Fuel	Novel fuel types, HALEU fuel
SSR-5	Disposal of Radioactive Waste	Novel wastes
SSG-1	Borehole Disposal Facilities for Radioactive Waste	No limitations
SSG-14	Geological Disposal Facilities for Radioactive Waste	WAC for novel wastes
SSG-23	The Safety Case and Safety Assessment for the Disposal of Radioactive Waste	No limitations
SSG-29	Near Surface Disposal Facilities for Radioactive Waste	WAC for novel wastes
SSG-31	Monitoring and Surveillance of Radioactive Waste Disposal Facilities	No limitations
GSG-16	Leadership, Management and Culture for Safety in Radioactive Waste Management	No limitations

No applicability considerations
Small number of applicability considerations or very small impact on the safety standard
Some applicability considerations or small impact on the safety standard

FIG. II.6. Safety standards for the waste and spent fuel management.

	IAEA Safety Standards Series publication	Main limitation
GSR Part 6	Decommissioning of Facilities	EIDs with centralized decommissioning
SSG-47	Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities	EIDs with centralized decommissioning, cogeneration
WS-G-5.1	Release of Sites from Regulatory Control on Termination of Practices	No limitations
WS-G-5.2	Safety Assessment for the Decommissioning of Facilities Using Radioactive Material	No limitations

No applicability considerations
Small number of applicability considerations or very small impact on the safety standard

FIG. II.7. Safety standards for the decommissioning of facilities.

IAEA Safety Standards Series publication		Main limitation
GSR Part 2	Leadership and Management for Safety	Ownership models
GS-G-3.1	Application of the Management System for Facilities and Activities	Supply chain, complex ownership, first of a kind design
GS-G-3.5	The Management System for Nuclear Installations	Supply chain, complex ownership

Small number of applicability considerations or very small impact on the safety standard

FIG. II.8. Safety standards for leadership and management for safety.

IAEA Safety Standards Series publication		Main limitation
GSR Part 4 (Rev. 1)	Safety Assessment for Facilities and Activities	Multiple modules, first of a kind design
SSG-2 (Rev. 1)	Deterministic Safety Analysis for Nuclear Power Plants	Non-WCRs
SSG-3	Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants	Non-WCRs, software and passive system reliability
SSG-4	Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants	Non-WCRs, functional containment approach, specific release possibilities for EIDs
SSG-25	Periodic Safety Review for Nuclear Power Plants	Impact of fuel dwell on periodicity of PSR
NS-G-2.13	Evaluation of Seismic Safety for Existing Nuclear Installations	TNPPs not placed at a site

Small number of applicability considerations or very small impact on the safety standard
Some applicability considerations or small impact on the safety standard
Numerous applicability considerations or more than a third of the safety standard impacted

FIG. II.9. Safety standards for safety assessment.

IAEA Safety Standards Series publication		Main limitation
GSR Part 7	Preparedness and Response for a Nuclear and Radiological Emergency	No limitations
GS-G-2.1	Arrangements for Preparedness for a Nuclear or Radiological Emergency	TNPPs, multiple modules
GSG-2	Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency	No limitations
GSG-11	Arrangements for the Termination of a Nuclear or Radiological Emergency	No limitations
GSG-14	Arrangements for Public Communication in Preparedness and Response for a Nuclear or Radiological Emergency	No limitations
SSG-65	Preparedness and Response for a Nuclear or Radiological Emergency Involving the Transport of Radioactive Material	TNPPs

No applicability considerations
Small number of applicability considerations or very small impact on the safety standard

FIG. II.10. Safety standards for emergency preparedness and response.

IAEA Safety Standards Series publication		Main limitation
GSR Part 1 (Rev. 1)	Governmental, Legal and Regulatory Framework for Safety	Foreign manufacture
GSG-12	Organization, Management and Staffing of the Regulatory Body for Safety	No limitations
GSG-13	Functions and Processes of the Regulatory Body for Safety	First of a kind, early engagement with the regulatory body, foreign manufacture
SSG-12	Licensing Process for Nuclear Installations	
GSG-6	Communication and Consultation with Interested Parties by the Regulatory Body	No limitations
GSG-9	Regulatory Control of Radioactive Discharges to the Environment	Underground EIDs
SSG-16 (Rev. 1)	Establishing the Safety Infrastructure for a Nuclear Power Programme	Limitations identified in tables of this appendix

No applicability considerations
Small number of applicability considerations or very small impact on the safety standard

FIG. II.11. Safety standards for the legal and regulatory framework for safety.

IAEA Safety Standards Series publication		Main limitation
SSR-6 (Rev. 1)	Regulations for the Safe Transport of Radioactive Material	TNPPs

Numerous applicability considerations or more than a third of the safety standard impacted

FIG. II.12. Safety standards for the safe transport of radioactive material.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Terms for Describing Advanced Nuclear Power Plants, IAEA Nuclear Energy Series No. NR-T-1.19, IAEA, Vienna (2023).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Advances in Small Modular Reactor Technology Developments — A Supplement to: IAEA Advanced Reactors Information System (ARIS), 2022 edn, IAEA, Vienna (2022).

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

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, ARIS — Advanced Reactors Information System,
https://aris.iaea.org/

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, "Research and development activities", Status of Molten Salt Reactor Technology, Technical Reports Series No. 489, IAEA, Vienna (in press) Sect. 5.

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Molten Salt Reactor Technology, Technical Reports Series No. 489, IAEA, Vienna (in press).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Applicability of Design Safety Requirements to Small Modular Reactor Technologies Intended for Near Term Deployment, IAEA-TECDOC-1936, IAEA, Vienna (2020).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Governmental, Legal and Regulatory Framework for Safety, IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), IAEA, Vienna (2016).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016).

[10] EUROPEAN COMMISSION, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3, IAEA, Vienna (2014).

[11] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016).

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste, IAEA Safety Standards Series No. GSR Part 5, IAEA, Vienna (2009).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Facilities, IAEA Safety Standards Series No. GSR Part 6, IAEA, Vienna (2014).

[14] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL CIVIL AVIATION ORGANIZATION, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, INTERPOL, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, PREPARATORY COMMISSION FOR THE COMPREHENSIVE NUCLEAR TEST-BAN-TREATY ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, UNITED NATIONS OFFICE FOR THE COORDINATION OF HUMANITARIAN AFFAIRS, WORLD HEALTH ORGANIZATION, WORLD METEOROLOGICAL ORGANIZATION, Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSR Part 7, IAEA, Vienna (2015).

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Communication and Consultation with Interested Parties by the Regulatory Body, IAEA Safety Standards Series No. GSG-6, IAEA, Vienna (2014).

[16] INTERNATIONAL ATOMIC ENERGY AGENCY, Organization, Management and Staffing of the Regulatory Body for Safety, IAEA Safety Standards Series No. GSG-12, IAEA, Vienna (2018).

[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Functions and Processes of the Regulatory Body for Safety, IAEA Safety Standards Series No. GSG-13, IAEA, Vienna (2018).

[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSR-1, IAEA, Vienna (2019).

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

[20] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. SSR-4, IAEA, Vienna (2017).

[21] INTERNATIONAL ATOMIC ENERGY AGENCY, Disposal of Radioactive Waste, IAEA Safety Standards Series No. SSR-5, IAEA, Vienna (2011).

[22] 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).

[23] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material, IAEA Safety Standards Series No. SSR-6 (Rev. 1), IAEA, Vienna (2018).

[24] INTERNATIONAL MARITIME ORGANIZATION, International Maritime Dangerous Goods (IMDG) Code, 2020 edn, IMO, London (2020).

[25] INTERNATIONAL ATOMIC ENERGY AGENCY, Site Survey and Site Selection for Nuclear Installations, IAEA Safety Standards Series No. SSG-35, IAEA, Vienna (2015).

[26] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-9 (Rev. 1), IAEA, Vienna (2022).
[27] INTERNATIONAL ATOMIC ENERGY AGENCY, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-18, IAEA, Vienna (2011).
[28] INTERNATIONAL ATOMIC ENERGY AGENCY, Volcanic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-21, IAEA, Vienna (2012).
[29] INTERNATIONAL ATOMIC ENERGY AGENCY, Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-3.6, IAEA, Vienna (2004).
[30] INTERNATIONAL ATOMIC ENERGY AGENCY, External Human Induced Events in Site Evaluation for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-79, IAEA, Vienna (2023).
[31] INTERNATIONAL ATOMIC ENERGY AGENCY, Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-3.2, IAEA, Vienna (2002).
[32] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Classification of Structures, Systems and Components in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-30, IAEA, Vienna (2014).
[33] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of the Reactor Core for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-52, IAEA, Vienna (2019).
[34] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-56, IAEA, Vienna (2020).
[35] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of the Reactor Containment and Associated Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-53, IAEA, Vienna (2019).
[36] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Electrical Power Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-34, IAEA, Vienna (2016).
[37] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Instrumentation and Control Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-39, IAEA, Vienna (2016).
[38] INTERNATIONAL ATOMIC ENERGY AGENCY, Human Factors Engineering in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-51, IAEA, Vienna (2019).
[39] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Auxiliary Systems and Supporting Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-62, IAEA, Vienna (2020).

[40] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Fuel Handling and Storage Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-63, IAEA, Vienna (2020).
[41] INTERNATIONAL ATOMIC ENERGY AGENCY, Protection Against Internal Hazards in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-64, IAEA, Vienna (2021).
[42] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Nuclear Installations Against External Events Excluding Earthquakes, IAEA Safety Standards Series No. SSG-68, IAEA, Vienna (2021).
[43] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Design for Nuclear Installations, IAEA Safety Standards Series No. SSG-67, IAEA, Vienna (2021).
[44] INTERNATIONAL ATOMIC ENERGY AGENCY, Equipment Qualification for Nuclear Installations, IAEA Safety Standards Series No. SSG-69, IAEA, Vienna (2021).
[45] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection Aspects of Design for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.13, IAEA, Vienna (2005).
[46] INTERNATIONAL ATOMIC ENERGY AGENCY, Construction for Nuclear Installations, IAEA Safety Standards Series No. SSG-38, IAEA, Vienna (2015).
[47] INTERNATIONAL ATOMIC ENERGY AGENCY, Commissioning for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-28, IAEA, Vienna (2014).
[48] INTERNATIONAL ATOMIC ENERGY AGENCY, Conduct of Operations at Nuclear Power Plants, IAEA Safety Standards Series No. SSG-76, IAEA, Vienna (2022).
[49] INTERNATIONAL ATOMIC ENERGY AGENCY, The Operating Organization for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-72, IAEA, Vienna (2022).
[50] INTERNATIONAL ATOMIC ENERGY AGENCY, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-70, IAEA, Vienna (2022).
[51] INTERNATIONAL ATOMIC ENERGY AGENCY, Modifications to Nuclear Power Plants, IAEA Safety Standards Series No. SSG-71, IAEA, Vienna (2022).
[52] INTERNATIONAL ATOMIC ENERGY AGENCY, Core Management and Fuel Handling for Nuclear Power Plants Safety Guide, IAEA Safety Standards Series No. SSG-73, IAEA, Vienna (2022).
[53] INTERNATIONAL ATOMIC ENERGY AGENCY, Criticality Safety in the Handling of Fissile Material, IAEA Safety Standards Series No. SSG-27 (Rev. 1), IAEA, Vienna (2022).
[54] INTERNATIONAL ATOMIC ENERGY AGENCY, Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-74, IAEA, Vienna (2022).
[55] INTERNATIONAL ATOMIC ENERGY AGENCY, Recruitment, Qualification and Training of Personnel for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-75, IAEA, Vienna (2022).
[56] INTERNATIONAL ATOMIC ENERGY AGENCY, Chemistry Programme for Water Cooled Nuclear Power Plants, IAEA Safety Standards Series No. SSG-13, IAEA, Vienna (2011).

[57] INTERNATIONAL ATOMIC ENERGY AGENCY, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-48, IAEA, Vienna (2018).
[58] INTERNATIONAL ATOMIC ENERGY AGENCY, Operating Experience Feedback for Nuclear Installations, IAEA Safety Standards Series No. SSG-50, IAEA, Vienna (2018).
[59] INTERNATIONAL ATOMIC ENERGY AGENCY, Protection Against Internal and External Hazards in the Operation of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-77, IAEA, Vienna (2022).
[60] INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Management Programmes for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-54, IAEA, Vienna (2019).
[61] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Conversion Facilities and Uranium Enrichment Facilities, IAEA Safety Standards Series No. SSG-5, IAEA, Vienna (2010).
[62] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Uranium Fuel Fabrication Facilities, IAEA Safety Standards Series No. SSG-6, IAEA, Vienna (2010).
[63] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Uranium and Plutonium Mixed Oxide Fuel Fabrication Facilities, IAEA Safety Standards Series No. SSG-7, IAEA, Vienna (2010).
[64] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Fuel Reprocessing Facilities, IAEA Safety Standards Series No. SSG-42, IAEA, Vienna (2017).
[65] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Fuel Cycle Research and Development Facilities, IAEA Safety Standards Series No. SSG-43, IAEA, Vienna (2017).
[66] INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, Occupational Radiation Protection, IAEA Safety Standards Series No. GSG-7, IAEA Vienna (2018).
[67] INTERNATIONAL ATOMIC ENERGY AGENCY, UNITED NATIONS ENVIRONMENT PROGRAMME, Radiation Protection of the Public and the Environment, IAEA Safety Standards Series No. GSG-8, IAEA, Vienna (2018).
[68] INTERNATIONAL ATOMIC ENERGY AGENCY, UNITED NATIONS ENVIRONMENT PROGRAMME, Regulatory Control of Radioactive Discharges to the Environment, IAEA Safety Standards Series No. GSG-9, IAEA, Vienna (2018).
[69] INTERNATIONAL ATOMIC ENERGY AGENCY, UNITED NATIONS ENVIRONMENT PROGRAMME, Prospective Radiological Environmental Impact Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSG-10, IAEA, Vienna (2018).
[70] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Concepts of Exclusion, Exemption and Clearance, IAEA Safety Standards Series No. RS-G-1.7, IAEA, Vienna (2004).
[71] INTERNATIONAL ATOMIC ENERGY AGENCY, Classification of Radioactive Waste, IAEA Safety Standards Series No. GSG-1, IAEA, Vienna (2009).

[72] INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership, Management and Culture for Safety in Radioactive Waste Management, IAEA Safety Standards Series No. GSG-16, IAEA, Vienna (2022).
[73] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. SSG-40, IAEA, Vienna (2016).
[74] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste from Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. SSG-41, IAEA, Vienna (2016).
[75] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste from the Use of Radioactive Material in Medicine, Industry, Agriculture, Research and Education, IAEA Safety Standards Series No. SSG-45, IAEA, Vienna (2019).
[76] INTERNATIONAL ATOMIC ENERGY AGENCY, Storage of Radioactive Waste, IAEA Safety Standards Series No. WS-G-6.1, IAEA, Vienna (2006).
[77] INTERNATIONAL ATOMIC ENERGY AGENCY, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste, IAEA Safety Standards Series No. GSG-3, IAEA, Vienna (2013).
[78] INTERNATIONAL ATOMIC ENERGY AGENCY, Storage of Spent Nuclear Fuel, IAEA Safety Standards Series No. SSG-15 (Rev. 1), IAEA, Vienna (2020).
[79] INTERNATIONAL ATOMIC ENERGY AGENCY, Borehole Disposal Facilities for Radioactive Waste, IAEA Safety Standards Series No. SSG-1, IAEA, Vienna (2009).
[80] INTERNATIONAL ATOMIC ENERGY AGENCY, Monitoring and Surveillance of Radioactive Waste Disposal Facilities, IAEA Safety Standards Series No. SSG-31, IAEA, Vienna (2014).
[81] INTERNATIONAL ATOMIC ENERGY AGENCY, Geological Disposal Facilities for Radioactive Waste, IAEA Safety Standards Series No. SSG-14, IAEA, Vienna (2011).
[82] INTERNATIONAL ATOMIC ENERGY AGENCY, The Safety Case and Safety Assessment for the Disposal of Radioactive Waste, IAEA Safety Standards Series No. SSG-23, IAEA, Vienna (2012).
[83] INTERNATIONAL ATOMIC ENERGY AGENCY, Near Surface Disposal Facilities for Radioactive Waste, IAEA Safety Standards Series No. SSG-29, IAEA, Vienna (2014).
[84] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. SSG-47, IAEA, Vienna (2018).
[85] INTERNATIONAL ATOMIC ENERGY AGENCY, Release of Sites from Regulatory Control on Termination of Practices, IAEA Safety Standards Series No. WS-G-5.1, IAEA, Vienna (2006).
[86] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for the Decommissioning of Facilities Using Radioactive Material, IAEA Safety Standards Series No. WS-G-5.2, IAEA, Vienna (2008).
[87] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

[88] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Nuclear Installations, IAEA Safety Standards Series No. GS-G-3.5, IAEA, Vienna (2009).

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

[90] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-3, IAEA, Vienna (2010).

[91] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-4, IAEA, Vienna (2010).

[92] INTERNATIONAL ATOMIC ENERGY AGENCY, Periodic Safety Review for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-25, IAEA, Vienna (2013).

[93] INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of Seismic Safety for Existing Nuclear Installations, IAEA Safety Standards Series No. NS-G-2.13, IAEA, Vienna (2009).

[94] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS OFFICE FOR THE COORDINATION OF HUMANITARIAN AFFAIRS, WORLD HEALTH ORGANIZATION, Arrangements for Preparedness for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GS-G-2.1, IAEA, Vienna (2007).

[95] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, PAN AMERICAN HEALTH ORGANIZATION, WORLD HEALTH ORGANIZATION, Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSG-2, IAEA, Vienna (2011).

[96] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL CIVIL AVIATION ORGANIZATION, INTERNATIONAL LABOUR OFFICE, INTERNATIONAL MARITIME ORGANIZATION, INTERPOL, OECD NUCLEAR ENERGY AGENCY, UNITED NATIONS OFFICE FOR THE COORDINATION OF HUMANITARIAN AFFAIRS, WORLD HEALTH ORGANIZATION, WORLD METEOROLOGICAL ORGANIZATION, Arrangements for the Termination of a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSG-11, IAEA, Vienna (2018).

[97] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL CIVIL AVIATION ORGANIZATION, INTERPOL, PREPARATORY COMMISSION FOR THE COMPREHENSIVE NUCLEAR-TEST-BAN TREATY ORGANIZATION, UNITED NATIONS OFFICE FOR OUTER SPACE AFFAIRS, Arrangements for Public Communication in Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSG-14, IAEA, Vienna (2020).

[98] INTERNATIONAL ATOMIC ENERGY AGENCY, Preparedness and Response for a Nuclear or Radiological Emergency Involving the Transport of Radioactive Material, IAEA Safety Standards Series No. SSG-65, IAEA, Vienna (2022).

[99] INTERNATIONAL ATOMIC ENERGY AGENCY, Licensing Process for Nuclear Installations, IAEA Safety Standards Series No. SSG-12, IAEA, Vienna (2010).

[100] 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).

[101] INTERNATIONAL ATOMIC ENERGY AGENCY, Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (2018 Edition), IAEA Safety Standards Series No. SSG-26 (Rev. 1), IAEA, Vienna (2022).

[102] INTERNATIONAL ATOMIC ENERGY AGENCY, Compliance Assurance for the Safe Transport of Radioactive Material, IAEA Safety Standards Series No. TS-G-1.5, IAEA, Vienna (2009).

[103] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for the Safe Transport of Radioactive Material, IAEA Safety Standards Series No. TS-G-1.4, IAEA, Vienna (2008).

[104] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection Programmes for the Transport of Radioactive Material, IAEA Safety Standards Series No. TS-G-1.3, IAEA, Vienna (2007).

[105] INTERNATIONAL ATOMIC ENERGY AGENCY, Schedules of Provisions of the IAEA Regulations for the Safe Transport of Radioactive Material (2018 Edition), IAEA Safety Standards Series No. SSG-33 (Rev. 1), IAEA, Vienna (2021).

[106] INTERNATIONAL ATOMIC ENERGY AGENCY, Format and Content of the Package Design Safety Report for the Transport of Radioactive Material, IAEA Safety Standards Series No. SSG-66, IAEA, Vienna (2022).

[107] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5), IAEA Nuclear Security Series No. 13, IAEA, Vienna (2011).

[108] The Structure and Content of Agreements Between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons, INFCIRC/153 (Corrected), IAEA, Vienna (1972).

[109] Model Protocol Additional to the Agreement(s) between State(s) and the International Atomic Energy Agency for the Application of Safeguards, INFCIRC/540 (Corrected), IAEA, Vienna (1998).

[110] INTERNATIONAL ATOMIC ENERGY AGENCY, Objective and Essential Elements of a State's Nuclear Security Regime, IAEA Nuclear Security Series No. 20, IAEA, Vienna (2013).
[111] INTERNATIONAL ATOMIC ENERGY AGENCY, Physical Protection of Nuclear Material and Nuclear Facilities (Implementation of INFCIRC/225/Revision 5), IAEA Nuclear Security Series No. 27-G, IAEA, Vienna (2018).
[112] INTERNATIONAL ATOMIC ENERGY AGENCY, Preventive and Protective Measures Against Insider Threats, IAEA Nuclear Security Series No. 8-G (Rev. 1), IAEA, Vienna (2020).
[113] INTERNATIONAL ATOMIC ENERGY AGENCY, International Legal Framework for Nuclear Security, IAEA International Law Series No. 4, IAEA, Vienna (2011).
[114] The Convention on the Physical Protection of Nuclear Material, INFCIRC/274/Rev. 1, IAEA, Vienna (1980).
[115] Amendment to the Convention on the Physical Protection of Nuclear Material, INFCIRC/274/Rev. 1/Mod. 1 (Corrected), IAEA, Vienna (2021).
[116] Convention on Early Notification of a Nuclear Accident, INFCIRC/335, IAEA, Vienna (1986).
[117] Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency, INFCIRC/336, IAEA, Vienna (1986).
[118] UNITED NATIONS, Security Council Resolution S/RES/1373, UNSC, New York (2001).
[119] UNITED NATIONS, Security Council Resolution S/RES/154, UNSC, New York (2004).
[120] UNITED NATIONS, International Convention for the Suppression of Acts of Nuclear Terrorism, United Nations General Assembly Resolution A/RES/59/290, UN, New York (2005).
[121] INTERNATIONAL ATOMIC ENERGY AGENCY, Code of Conduct on the Safety and Security of Radioactive Sources, IAEA/CODEOC/2004, IAEA, Vienna (2004).
[122] THE WHITE HOUSE, Communiqué of the Washington Nuclear Security Summit, Office of the Press Secretary, Washington, DC (2010).
[123] GOVERNMENT OF THE REPUBLIC OF KOREA, Seoul Communiqué, 2012 Seoul Nuclear Security Summit, Seoul (2012).
[124] GOVERNMENT OF THE NETHERLANDS, The Hague Nuclear Security Summit Communiqué, The Hague (2014).
[125] THE WHITE HOUSE, Nuclear Security Summit 2016 Communiqué, Office of the Press Secretary, Washington, DC (2016).
[126] SMALL MODULAR REACTORS REGULATORS' FORUM, Working Group on Manufacturing, Construction, Commissioning and Operation (2021), https://www.iaea.org/sites/default/files/21/06/working_group_on_manufacturing_construction_commissioning_and_operation_phase_2_report.pdf
[127] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Glossary: Terminology Used in Nuclear Safety and Radiation Protection, 2018 Edition, IAEA, Vienna (2019).

[128] INTERNATIONAL ATOMIC ENERGY AGENCY, Technology Roadmap for Small Modular Reactor Deployment, IAEA Nuclear Energy Series No. NR-T-1.18, IAEA, Vienna (2021).
[129] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Medical, Industrial and Research Facilities, IAEA Safety Standards Series No. SSG-49, IAEA, Vienna (2019).
[130] INTERNATIONAL MARITIME ORGANIZATION, International Code for the Safe Carriage of Packaged Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Wastes on Board Ships (INF Code), Resolution MSC.88(71), IMO, London (1999).
[131] INTERNATIONAL MARITIME ORGANIZATION, SOLAS, Consolidated Edition 2014: Consolidated Text of the International Convention for the Safety of Life at Sea, 1974, and Its Protocol of 1988: Articles, Annexes and Certificates, IMO, London (2014).
[132] Treaty on the Non-Proliferation of Nuclear Weapons, INFCIRC/140, IAEA, Vienna (1970).
[133] BECKER, P., et al., "Implementation of safeguards measures at the high temperature gas-cooled reactor pebble-bed module (HTR-PM) in China and proposed safeguards by design for units to be exported to other States", Proc. IAEA Symposium on International Safeguards, Vienna, 2018 (abstract), https://www.iaea.org/sites/default/files/19/07/cn-267-book-of-abstracts.pdf
[134] INTERNATIONAL ATOMIC ENERGY AGENCY, Security of Nuclear Material in Transport, IAEA Nuclear Security Series No. 26-G, IAEA, Vienna (2015).
[135] INTERNATIONAL ATOMIC ENERGY AGENCY, Security of Radioactive Material in Transport, IAEA Nuclear Security Series No. 9-G (Rev. 1), IAEA, Vienna (2020).
[136] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Security Recommendations on Radioactive Material and Associated Facilities, IAEA Nuclear Security Series No. 14, IAEA, Vienna (2011).
[137] INTERNATIONAL ATOMIC ENERGY AGENCY, Handbook on the Design of Physical Protection Systems for Nuclear Material and Nuclear Facilities, IAEA Nuclear Security Series No. 40-T, IAEA, Vienna (2021).
[138] INTERNATIONAL ATOMIC ENERGY AGENCY, Computer Security Techniques for Nuclear Facilities, IAEA Nuclear Security Series No. 17-T (Rev. 1), IAEA, Vienna (2021).
[139] INTERNATIONAL ATOMIC ENERGY AGENCY, Computer Security for Nuclear Security, IAEA Nuclear Security Series No. 42-G, IAEA, Vienna (2021).

ABBREVIATIONS

AOO	anticipated operational occurrence
ATF	accident tolerant fuel
BWR	boiling water reactor
CCF	common cause failure
DBA	design basis accident
DEC	design extension condition
DHR	decay heat removal
DiD	defence in depth
DSA	deterministic safety analysis
EAL	emergency action level
EID	evolutionary and innovative design
EPC	emergency preparedness category
EPD	emergency planning distance
EPR	emergency preparedness and response
EPZ	emergency planning zone
HALEU	high assay low enriched uranium
HSI	human–system interface
HTGR	high temperature gas cooled reactor
HVAC	heating, ventilation and air-conditioning
I&C	instrumentation and control
IMO	International Maritime Organization
LEU	low enriched uranium
LFR	lead cooled fast reactor
LOCA	loss of coolant accident
MBA	material balance area
MCR	main control room
MOX	mixed oxide
MSR	molten salt reactor
NDA	non-destructive analysis
NPP	nuclear power plant
OLCs	operational limits and conditions
PBR	pebble bed reactor
PHWR	pressurized heavy water reactor
PIE	postulated initiating event
PPS	physical protection system
PSA	probabilistic safety analysis
PSR	periodic safety review
PUREX	plutonium uranium extraction reduction

PWR	pressurized water reactor
R&D	research and development
RCS	reactor coolant system
SBD	safeguards by design
SFR	sodium cooled fast reactor
SGTR	steam generator tube rupture
SMR	small modular reactor
SNF	spent nuclear fuel
SSAC	state system of accounting for and control of nuclear material
SSCs	structures, systems and components
TNPP	transportable nuclear power plant
TRISO	tristructural isotropic
UHS	ultimate heat sink
WAC	waste acceptance criteria
WCR	water cooled reactor

CONTRIBUTORS TO DRAFTING AND REVIEW

Aamir, M.A.	Ontario Power Generation, Canada
Abrosimov, A.	Afrikantov OKBM JSC, Russian Federation
Adam, N.	Office for Nuclear Regulation, United Kingdom
Adkins, H.	Pacific Northwest National Laboratory, United States of America
Agarwal, K.	International Atomic Energy Agency
Albert, M.	Electric Power Research Institute, United States of America
Alemberti, A.	Consultant, Italy
Allsopp, G.	Office for Nuclear Regulation, United Kingdom
Bamber, M.	Office for Nuclear Regulation, United Kingdom
Barley, D.	Office for Nuclear Regulation, United Kingdom
Barrett, A.	Nuclear Regulatory Commission, United States of America
Bedrossian, S.	Ontario Power Generation, Canada
Bennett, D.G.	International Atomic Energy Agency
Bertrand, R.	Institute for Radiological Protection and Nuclear Safety, France
Bilic Zabric, T.	International Atomic Energy Agency
Boyd, C.	Office for Nuclear Regulation, United Kingdom
Brown, J.	International Atomic Energy Agency
Buchanan, T.	Nuclear Regulatory Commission, United States of America
Calle Vives, P.	International Atomic Energy Agency
Carluec, B.	Consultant, France

Charkas, H.	Electric Power Research Institute, United States of America
Choi, J.	Consultant, United States of America
Coman, O.	International Atomic Energy Agency
Courtin, E.	Framatome, France
Crossland, I.	Consultant, United Kingdom
Curtis, R.	Office for Nuclear Regulation, United Kingdom
De Vos, M.	Canadian Nuclear Safety Commission, Canada
Delfini, G.	Authority for Nuclear Safety and Radiation Protection, Netherlands
Dengcheng, S.	China Nuclear Power Technology Research Institute, China
Devita, M.	Office for Nuclear Regulation, United Kingdom
Diaz, M.	Nuclear Regulatory Commission, United States of America
Donnelly, P.	Office for Nuclear Regulation, United Kingdom
Doronkova, D.	Afrikantov OKBM JSC, Russian Federation
Duchac, A.	International Atomic Energy Agency
Franovich, M.	Nuclear Regulatory Commission, United States of America
Gajdos, M.	International Atomic Energy Agency
Galitskikh, V.	Afrikantov OKBM JSC, Russian Federation
Geupel, S.	International Atomic Energy Agency
Graham, S.	Office for Nuclear Regulation, United Kingdom
Guzman Lopez-Ocon, O.	International Atomic Energy Agency
Halhead, S.	Office for Nuclear Regulation, United Kingdom
Hardwood, C.	Canadian Nuclear Safety Commission, Canada

Hart, M.	Nuclear Regulatory Commission, United States of America
Haseen, F.	Ontario Power Generation, Canada
Heath, M.	Nuclear Regulatory Commission, United States of America
Henneke, D.	General Electric, United States of America
Hernandez, R.	Nuclear Regulatory Commission, United States of America
Hilton, M.	Office for Nuclear Regulation, United Kingdom
Hoellman, J.	Nuclear Regulatory Commission, United States of America
Holcomb, D.E.	Oak Ridge National Laboratory, United States of America
Horvath, K.	International Atomic Energy Agency
Illizastigui Perez, F.	Inversiones Gamma S.A., Cuba
Iyengar, A.	Department of Energy, United States of America
Jansen, R.	Authority for Nuclear Safety and Radiation Protection, Netherlands
Johnson, L.	Office for Nuclear Regulation, United Kingdom
Kameswaran, R.	Canadian Nuclear Safety Commission, Canada
Katona, T.J.	Paks Nuclear Power Plant, Hungary
Kavanagh, K.	Nuclear Regulatory Commission, United States of America
Kent, M.	Canadian Nuclear Safety Commission, Canada
Kirkin A.	Scientific and Engineering Centre for Nuclear and Radiation Safety, Russian Federation
Kovacic, D.	Oak Ridge National Laboratory, United States of America
Krepel, J.	Paul Scherrer Institute, Switzerland

Kubanova, I.	International Atomic Energy Agency
Kubo, S.	Japan Atomic Energy Agency, Japan
Kuryndin A.	Scientific and Engineering Centre for Nuclear and Radiation Safety, Russian Federation
Lafleur, A.	Los Alamos National Laboratory, United States of America
Lankin, M.	International Atomic Energy Agency
Lepouze, B.	International Atomic Energy Agency
Li, F.	Institute of Nuclear and New Energy Technology, China
Li, J.	Institute of Nuclear and New Energy Technology, China
Lipar, M.	Consultant, Slovakia
Lisbona, D.	Office for Nuclear Regulation, United Kingdom
Ljubenov, V.	International Atomic Energy Agency
Lu, S.	Nuclear Regulatory Commission, United States of America
Ma, J.	International Atomic Energy Agency
Ma, T.	Institute of Nuclear and New Energy Technology, China
Maheras, S.	Pacific Northwest National Laboratory, United States of America
Majeed, T.	International Atomic Energy Agency
Malev, V.	Afrikantov OKBM JSC, Russian Federation
Martin, G.	Ontario Power Generation, Canada
Martinez-Guridi, G.	International Atomic Energy Agency
Matharu, G.	Nuclear Regulatory Commission, United States of America
Mazza, J.	Nuclear Regulatory Commission, United States of America

McDowall, B.	Office for Nuclear Regulation, United Kingdom
Misak, J.	Consultant, Czech Republic
Monti, S.	International Atomic Energy Agency
Morgan, S.	International Atomic Energy Agency
Murphy, P.	Office for Nuclear Regulation, United Kingdom
Nestoroska Madjunarova, S.	International Atomic Energy Agency
Nie, J.	Institute of Nuclear and New Energy Technology, China
Nitsche, F.	Federal Office for Radiation Protection, Germany
Nolan, R.	Nuclear Regulatory Commission, United States of America
Nünighoff, K.	Gesellschaft für Anlagen- und Reaktorsicherheit, Germany
Ohashi, H.	Japan Atomic Energy Agency, Japan
Parada Iturria, F.	International Atomic Energy Agency
Pázmándi, T.	Centre for Energy Research, Hungary
Phan, H.	Nuclear Regulatory Commission, United States of America
Plummer, D.	Office for Nuclear Regulation, United Kingdom
Poghosyan, S.	International Atomic Energy Agency
Poulat, B.	Consultant, France
Prescott, P.	Nuclear Regulatory Commission, United States of America
Ray, S.	Nuclear Regulatory Commission, United States of America
Renda, G.	European Commission – Joint Research Centre, Italy
Rivera, R.	Nuclear Regulatory Commission, United States of America

Rodriguez, V.	Nuclear Regulatory Commission, United States of America
Rovny, J.	International Atomic Energy Agency
Rubenstone, J.	Nuclear Regulatory Commission, United States of America
Rudland, D.	Nuclear Regulatory Commission, United States of America
Salnikova, N.	Afrikantov OKBM JSC, Russian Federation
Santini, M.	International Atomic Energy Agency
Schrader, E.	Nuclear Regulatory Commission, United States of America
Schultz, S.	Consultant, United States of America
Seymour, J.	Nuclear Regulatory Commission, United States of America
Shahrokhi, F.	Framatome, United States of America
Sinegribov, S.	Scientific and Engineering Centre for Nuclear and Radiation Safety, Russian Federation
Siwy, A.	Nuclear Regulatory Commission, United States of America
Sorokin, D.	Scientific and Engineering Centre for Nuclear and Radiation Safety, Russian Federation
Stattel, R.	Nuclear Regulatory Commission, United States of America
Stenhoff, P.	Office for Nuclear Regulation, United Kingdom
Stephani, F.	International Atomic Energy Agency
Stutzcage, E.	Nuclear Regulatory Commission, United States of America
Subki, H.	International Atomic Energy Agency
Tate, R.	Office for Nuclear Regulation, United Kingdom

Thorington, R.	Office for Nuclear Regulation, United Kingdom
Toth, C.	Consultant, Hungary
Tyndall, D.	Ontario Power Generation, Canada
Valliere, N.	Nuclear Regulatory Commission, United States of America
Van Wert, C.	Nuclear Regulatory Commission, United States of America
Vettori, R.	Nuclear Regulatory Commission, United States of America
Videla, M.L.	International Atomic Energy Agency
Villalibre, P.	International Atomic Energy Agency
Wagage, H.	Nuclear Regulatory Commission, United States of America
Walter, A.	TÜV Rheinland Industrie Service, Germany
Wang, H.	Institute of Nuclear and New Energy Technology, People's Republic of China
White, D.	Nuclear Regulatory Commission, United States of America
Whitlock, J.	International Atomic Energy Agency
Wielenberg, A.	Gesellschaft für Anlagen- und Reaktorsicherheit, Germany
Zhang, Y.	Institute of Nuclear and New Energy Technology, People's Republic of China
Zhao, J.	Institute of Nuclear and New Energy Technology, People's Republic of China
Zheng, Y.	Institute of Nuclear and New Energy Technology, People's Republic of China

Consultants Meetings

Vienna, Austria: 15–19 March 2021; 10–19 May 2021;
12–16 July 2021; 19–21 October 2021

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

23-01543E-T

In response to an increase in global activities related to non-water cooled reactors and small modular reactors, this Safety Report documents areas of novelty of these technologies in comparison with the current fleet of reactors. The impact of these areas of novelty on the applicability and completeness of the IAEA safety standards is assessed in the publication. Gaps and areas for additional consideration are identified. The review undertaken to develop this report encompassed the safety standards related to the lifetime of these reactor technologies. The publication also considers the interface between safety, security and safeguards in the design of these technologies.

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA