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Technical Basis for the Emergency Planning Zone for Small Modular Reactors

Final Report of a Coordinated Research Project

TECHNICAL BASIS
FOR THE EMERGENCY PLANNING ZONE
FOR SMALL MODULAR REACTORS

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TECHNICAL BASIS FOR THE EMERGENCY PLANNING ZONE FOR SMALL MODULAR REACTORS

FINAL REPORT OF A COORDINATED RESEARCH PROJECT

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2026

For further information on this publication, please contact:

Nuclear Power Technology Development Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
Email: Official.Mail@iaea.org

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FOREWORD

The IAEA supports the efforts of Member States to harness nuclear energy in the transition to net zero emissions. This support includes technical expertise for decarbonizing electricity production, as well as hard to abate sectors such as industry and transport. Interest among Member States in small modular reactors (SMRs), including microreactors, is increasing. Significant institutional and industrial efforts are being directed towards facilitating the development and early deployment of SMRs to contribute to the transition to net zero. Two nuclear power plants with SMRs are already in operation, and more than 80 SMR designs are at various stages of development across major technology lines worldwide; eight SMR units are currently under construction, and construction of additional units is expected to begin by 2030 in several countries. With a typical electrical output of up to 300 MW(e) per reactor unit, SMRs have distinct design, safety and siting features, as well as applications. Member States have requested that the IAEA organize coordinated research activities to assess, formulate and implement adequate emergency preparedness and response arrangements, including determination of the size of the emergency planning zone for SMRs. Emergency preparedness and response is a key prerequisite in the development of a safe national infrastructure for nuclear power.

In response to Member State requests, the IAEA launched the Coordinated Research Project on the Development of Approaches, Methodologies and Criteria for Determining the Technical Basis for Emergency Planning Zone for Small Modular Reactor Deployment (2018–2021). The four year project was established as an interdepartmental collaboration jointly funded by the Department of Nuclear Energy and the Department of Nuclear Safety and Security. Its main objective was to examine approaches and methodologies for determining the need for off-site emergency preparedness and response, including the emergency planning zone and its size for SMRs, taking into account the enhanced safety features of SMRs, as well as design specific and site specific features, in the context of potential deployment in participating Member States.

The coordinated research project participants comprised 19 leading research institutions and organizations: 18 from 13 Member States — Argentina, Canada, China, Finland, Indonesia, Israel, Japan, the Republic of Korea, Pakistan, Saudi Arabia, Tunisia, the United Kingdom and the United States of America — and one international organization, the European Commission's Joint Research Centre. This publication presents the outcomes of the various activities undertaken and the extensive discussions held during the research coordination meetings, and summarizes the information provided by the chief scientific investigators and their teams over the four year project period.

The IAEA officers responsible for this publication were F. Stephani of the Incident and Emergency Centre and H. Subki of the Division of Nuclear Power.

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1. INTRODUCTION

1.1. BACKGROUND

Small modular reactors (SMRs), with electrical power of up to 300 MW per unit module, have specific design, safety and siting features, as well as applications. These specificities need to be properly assessed to formulate and plan for adequate emergency preparedness and response (EPR) arrangements, in particular the size of the emergency planning zones (EPZs). The EPR is one of the 19 elements in the development of a national infrastructure for nuclear power, while the EPZ is one of the specific regulatory aspects to enable SMR deployment in diverse siting niches.

The main objective of coordinated research project (CRP) I31019, Development of Approaches, Methodologies and Criteria for Determining the Technical Basis for Emergency Planning Zone for Small Modular Reactor Deployment (2018–2021) (hereafter referred to as 'the CRP') was to discuss approaches and methodologies for determining the need for off-site EPR, including EPZs and their associated sizes for SMRs, taking into account the claimed enhanced safety performance of SMRs, as well as design specific and site specific features, for their potential deployment in the CRP participants' countries. EPZs consist of the precautionary action zone (PAZ) and the urgent protective action planning zone (UPZ). In the PAZ and UPZ, arrangements are made to take precautionary and urgent protective actions in the event of a nuclear or radiological emergency, to avoid or minimize severe deterministic effects off the site and to avert doses off the site, respectively.

There is strong interest in SMRs in Member States with operating nuclear power plants (NPPs) and embarking countries. Significant institutional and industrial efforts are being devoted to facilitating their development and early deployment. With this increase in global efforts and activities, the IAEA has received requests from Member States for support in increasing awareness of these innovative technologies, including their benefits and challenges.

Many embarking countries, including those with developing economies, have expressed an interest in deploying SMRs in their future energy portfolio. Some of the embarking countries have specific siting characteristics that might need specific attention with respect to EPR. Examples of potential markets for SMRs could include archipelago countries in Southeast Asia with isolated population areas and limited grids that might need distributed generation. In North America, SMRs are options for replacing ageing coal power plants as well as for providing independent power supply applications to support mining and remote communities.

Origin of this Coordinated Research Project

During the meetings of the IAEA Standing Advisory Group on Nuclear Energy (SAGNE) held in 2016 and 2017, the members of SAGNE discussed the claims and design expectations of the SMR technology developers and the potential users/utilities regarding specific features to obtain a reduced EPZ size. Questions were then raised concerning the circumstances in which, or the bases upon which, an EPZ size could be reduced — if any. Through different IAEA meetings regarding SMRs, participants from Member States expressed their wish to see the IAEA provide guidance on determining the technical basis for EPZs for SMRs through a CRP. The IAEA engaged in further planning in mid-2017 to launch a CRP that can provide a forum for R&D and technical exchange for Member States to address aspects of EPR specific to SMR deployment, particularly the size of EPZs.

The SMR Regulators' Forum was created in March 2015. It facilitates discussions among Member States and other relevant stakeholders to share SMR regulatory knowledge and experience.

In Pilot Project Report: Considering the Application of a Graded Approach, Defence-in-Depth and Emergency Planning Zone Size for Small Modular Reactors (January 2018) [1], the SMR Regulators' Forum concluded that:

(a) "Until the proven-ness has been established, it is difficult to claim credits for those features in a safety proposal because uncertainties need to be addressed and factored into the safety demonstration."
(b) "For DiD level 5, the DiD WG is in agreement with the NEA statement that, no matter how much other levels might be strengthened, effective emergency arrangements and other responses are essential to cover the unexpected."
(c) "The practical elimination concept should not be used to justify omission of a complete DiD level."
(d) "As for large reactors, PSAs should be used for SMRs to complement the deterministic approach on which the design relies first."

In Pilot Project Report: Report from Working Group on Defence-in-Depth (January 2018) [2], the SMR Regulators' Forum concluded that:

> "According to the limited publicly available information on SMR designs, the WG observed that "multi-modules" could not be considered as equivalent to "multi-units", as with large reactors. Further, such concepts were not well defined for SMRs. For instance, the "module" might or might not be autonomous and does not include individual safety systems and safety support systems such as separate heat sink. It was observed in some designs that the control room, reactor building and ultimate heat sink, as examples, can be common to several modules. In addition, some SMRs might use a single confinement common to several modules."

Lastly, in its Phase 2 Summary Report (June 2021) [3], the SMR Regulators' Forum concluded that:

> "The presence of multiple modules/units at the site could exacerbate challenges that the plant personnel would face during an accident. The events and consequences of an accident at one unit might affect the accident progression or hamper accident management activities at the neighbouring unit".

The IAEA Department of Nuclear Safety and Security and its Incident and Emergency Centre have been involved in EPR and EPZ considerations for advanced reactors, including SMRs. IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [4] is the key reference for discussion of the subject. This consensus publication establishes requirements for the goals of EPR, hazard assessment, emergency preparedness categories, protection strategy and protective actions, and EPZs.

1.2. OBJECTIVE

The main objective of this publication is to provide approaches, methodologies and criteria for determining the technical basis for EPR arrangements, focusing on the size of the EPZ for SMR deployments.

The specific objectives of the CRP were as follows:

(a) Formulate criteria on the events and technical aspects to be considered for defining EPR arrangements for SMRs, focusing on EPZ sizing. This should be based on research results and the implementation of defence in depth in the design of SMRs, including small power, smaller source term, increased safety margin, enhanced engineered safety systems, smaller and slower fission product release, and consequent reduced potential for radiation exposure to the population in the vicinity of the plant.
(b) Develop approaches and methodologies that enable the safety features of SMRs to be related to the extent of the off-site arrangements required, particularly the size of the EPZs, by comparing the design and site-specific technical bases to be provided by SMR developers, nuclear regulators, emergency planners and users/utilities.
(c) Provide a suitable technical basis as input for the development of additional IAEA technical guidance (EPR series report) on EPR arrangements for SMRs. Also, additional input for new guidance regarding source term definition and assessment could be derived on that basis, as appropriate.

1.3. SCOPE

The CRP was launched to study the EPZ, as applicable, of any SMR designs defined in general as advanced or newer generation reactors that generate typically up to 300 MW(e) whose systems and components can be shop fabricated as modules to reduce construction time and thus reduce the cost. The agreed focus was on land based SMRs with one research topic on the case for modular high temperature gas cooled reactors (HTGR). There was no restriction in the choice of methodology for developing the EPZ. The work conducted put an emphasis on accident phenomenology, accident scenarios, source term calculation, atmospheric dispersion and dose projection.

1.4. STRUCTURE

The publication has the following structure:

— Section 1 provides an introduction covering the background, scope and objectives of the CRP, as well as the structure of the publication;
— Section 2 provides general information on SMR technologies, as well as information specific to SMR's fission product barriers and associated safety systems;
— Section 3 provides information on the existing guidance from the IAEA and Member States (presented by the organizations participating in the CRP) for the determination of EPZs;

— Section 4 covers the technical steps for the determination of EPZs for SMRs, including accident sequences, source term calculations, atmospheric dispersion and dose projections, the main uncertainties, and findings on and discussion of EPZ sizes;
— Section 5 provides a summary of and a conclusion concerning the different methodologies that have been followed by the participant organizations (deterministic, probabilistic, best estimate, conservative) to select and model accident sequences, including accident on-site phenomenology and off-site consequences, as well as the strengths and limitations of these methodologies;
— Section 6 addresses the open issues and suggests identified activities to be conducted by the IAEA and/or in Member States.

2. CHARACTERIZATION OF SMALL MODULAR REACTORS

2.1. GENERAL AND COMMON CHARACTERISTICS OF SMALL MODULAR REACTORS

As an emerging technology, SMRs are advanced nuclear reactors, typically with a power capacity of up to 300 MW(e), whose structures, systems and components (SSCs) can be factory built and then transported as modules to sites for installation, minimizing on-site works. SMRs are under different phases of development and deployment for major types of reactor technologies; for example, water cooled reactors, HTGRs, liquid metal cooled and gas cooled reactors with fast neutron spectrum, molten salt reactors (MSRs) and microreactors.

2.1.1. Core inventory of small modular reactors

For the majority of SMR designs, the maximum power levels and associated dimensions are more flexible in terms of manufacturing and transportability, which are key features for SMRs to enable serial production and timely deployment. At lower power levels the core inventory, that is the quantity of fission products, activation products and actinides in the reactor core, is smaller. Because the actinide content does not scale directly with power, it should be quantified separately rather than inferred from the power level. The smaller core inventory means that there are fewer radionuclides present that could be released into the environment during an accident.

2.1.2. Modularization, serial factory production of modules and multi-module plant

Modularization is the proess by which SMRs are designed and assembled from smaller building blocks, or modules. SMR modules can be constructed in a factory and then transported to a site, making parallel construction activities possible and improving productivity. This technique aims at standardization of components to the extent practicable, factory fabrication and more simplified on-site installation of preassembled modules.

Factory built modules, as opposed to the stick built construction currently used for large scale NPPs, typically have been designed to be all inclusive with regard to mechanical, electrical and instrumentation features. They can be assembled in a more controlled plant environment by fully trained labour, using the same materials, codes and standards for each subsequent module.

Through modularization technology, SMRs target at the economies of serial production with shorter installation or construction times.

A combination of small power, design simplification and modularization to the nuclear island and turbine island are expected to result in reduced physical footprints and reduced capital investment. Hence, SMRs might be more easily sited in locations not possible for larger NPPs.

Smaller footprints are expected to enable SMRs to have flexibility in siting and to allow them to be tailored to the energy needs of regional or industrial clusters. The possibility of siting SMRs in remote, off-grid communities enhances the access of such communities to cleaner electricity and heat.

In synergy with other available alternatives, SMRs can be considered as a future flexible energy source for generating electricity and industrial process heat, particularly for countries and/or sites where large NPPs are not viable. Adopting inherent and passive safety features, the SMR design features might help decrease accident frequency and the associated consequences, and slow accident kinetics.

2.2. DIFFERENT TECHNOLOGIES USED IN SMALL MODULAR REACTORS

There are several ways of categorizing SMR technologies. One of them is the combined categorization by coolant-type, neutron spectrum and siting approach, as follows:

2.2.1. Water cooled land based SMRs

This category includes the well known light water reactor (LWR) and heavy water reactor (HWR) technologies. These designs take advantage of mature technology as most of the 'large' NPPs in operation consist of water cooled reactors. The designs include integral pressurized water reactors (PWRs), compact PWRs, loop type PWRs, pressurized heavy water reactors (PHWRs), boiling water reactors (BWRs) and pool type reactors for district heating.

2.2.2. Marine based SMRs

This category includes concepts that can be deployed in a marine environment, either as a barge mounted floating power units or immersible underwater power units. This subcategory provides many flexible deployment options. Near-term deployable designs are of water cooled reactors. Current developments of marine-based SMRs are also based non-water cooled reactor technologies, including MSRs and lead-bismuth cooled reactors with fast neutron spectrum. This technology will be addressed under the new initiative, Atomic Technologies Licensed for Applications at Sea (ATLAS), which may also involve considerations related to EPR and EPZ.

2.2.3. Modular high temperature gas cooled SMRs

This category includes HTGRs under development and in operation. HTGRs could provide high temperature heat (≥750°C) that can be utilized for more efficient electricity generation and a variety of industrial applications, as well as cogeneration. This category also includes HTGR test reactors, two of which have already been in operation for technology testing purposes for over 20 years.

2.2.4. Liquid metal cooled fast neutron spectrum SMRs

This category combines fast neutron spectrum with liquid metal coolants, including sodium, pure lead and lead–bismuth eutectic. Tangible advances in technological development and deployment for SMRs in this category have been made.

2.2.5. Molten salt fuelled and/or cooled SMRs

This category includes SMR designs from both molten salt fuelled and cooled advanced reactor technologies, which are also one of the six generation IV designs. MSRs would likely offer many advantages, including enhanced safety due to the inherent properties of the molten salt, a low pressure, single phase coolant system that eliminates the need for large containment, a high temperature system that results in high efficiency and a flexible fuel cycle.

2.2.6. Microreactors — from the above technologies and heat pipes

An unprecedented developmental trend emerged in the form of very small SMRs designed to generate electrical power of typically up to 10 MW(e). Different types of coolant, including light water, helium, molten salt and liquid metal, are adopted by microreactors. Heat pipes are another proposed cooling system option. Microreactors serve future niche electricity and heat markets, such as powering microgrids and remote off-grid areas, restoring power quickly in communities affected by natural disasters and supporting faster restoration of critical services (e.g. hospitals, water supply) and seawater desalination.

Another approach to classifying SMR technologies is outlined in the IAEA Safety Report Series No. 123, which addresses the applicability of IAEA Safety Standards to non-water-cooled reactors and small modular reactors [5].

In conducting their work in the CRP, some participants considered the following particular SMR technologies when researching the technical basis for determining EPZ size:

- Conventional two loop PWRs — 100 MW(e);
- Integral PWRs with forced convection — 100 MW(e), 125 MW(e) and 200 MW(e), respectively;
- Integral PWRs with natural circulation — 30 MW(e);
- Modular high temperature gas cooled reactors — 210 MW(e);
- Sodium cooled fast neutron spectrum.

However, some research contributions to the CRP addressing SMRs were presented generally and not with respect to a specific SMR technology.

2.3. FISSION PRODUCT BARRIERS AND ASSOCIATED SAFETY SYSTEMS FOR SMRs

According to para. 2.14 of IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [6]:

> “A relevant aspect of the implementation of defence in depth for a nuclear power plant is the provision in the design of a series of physical barriers, as well as a combination of active, passive and inherent safety features that contribute to the effectiveness of the physical barriers in confining radioactive material at specified locations. The number of barriers that will be necessary will depend upon the initial source term in terms of the amount and isotopic composition of radionuclides, the effectiveness of the individual barriers, the possible internal and external hazards, and the potential consequences of failures.”

This specific safety requirement applies in particular to water cooled type SMRs but also applies to many other major lines of technology.

2.3.1. Water cooled type SMRs

Barriers and associated safety systems for loop type PWRs, PHWRs and BWRs are well known. The combination of fuel matrix, cladding and reactor pressure vessel (RPV) that connects with the associated reactor coolant system forms a reactor coolant pressure boundary (RCPB). The RCPB is enclosed within a containment structure or building. This structure scheme typically comprises barriers in water cooled type SMRs. In this subsection, novel features in integral type SMRs are presented.

Depending on the design products, the application of fission product barriers to satisfy defence in depth varies. In the integral PWR designs, as with the loop type PWRs, the RPV holds the individual fuel assemblies and the control rods. However, the RPV in the integral designs houses also the steam generators and the pressurizer system. Thus, the entire reactor coolant inventory is contained within the RPV and large bore coolant circulation piping can be eliminated. Functionally, the RPV provides one of several safety barriers to fission product release, supporting the control rods and the reactor vessel internals, which support the reactor fuel and direct coolant flow within the vessel. One of the novelties for nuclear safety is to place the containment structure that houses the RPV in a common reactor pool that provides an additional source of cooling and barrier to fission product release.

2.3.2. Modular HTGR type SMRs

The major safety goal for modular HTGRs is that significant fuel failure results neither from postulated design basis accidents (DBAs) nor, as far as possible, from other events (e.g. design extension conditions (DECs)). This safety goal drives the design features that define modular HTGRs. Key fission product barriers for the reactor consist of the following.

2.3.2.1. Coated fuel particles

Fuel elements with coated particles serve as the first barrier. The fuel elements used have been tested extensively as part of the demonstration and shown to be capable of retaining fission products and other radionuclides within the coated particles under temperatures of ~1600°C.

2.3.2.2. Primary pressure boundary

The second barrier is the primary pressure boundary, which consists of the pressure vessels of the primary components (e.g. RPV, steam generator pressure vessel, hot gas duct pressure vessel).

2.3.2.3. Vented low pressure confinement

The low pressure vented confinement consists of auxiliary systems such as subatmosphere ventilation and filters. It is designed to to vent the helium during a blowdown. The filters are designed to mitigate the release of radioactive materials during that blowdown. After the blowdown, the radioactive materials escape via leakage.

2.3.3. Sodium cooled type SMRs with fast neutron spectrum

The typical barriers for a sodium cooled fast reactors (SFR) comprise fuel cladding tubes, a reactor vessel with a roof slab and a containment system that houses the reactor coolant boundary and cover gas boundary.

3. APPROACHES FOR THE DETERMINATION OF EMERGENCY PLANNING ZONES AND EMERGENCY PLANNING DISTANCES

3.1. IAEA SAFETY STANDARDS AND EXISTING GUIDANCE ON EPZs AND EPDs

3.1.1. Introduction

In 2021, the IAEA's Incident and Emergency Centre conducted a review of the EPR-related safety standards for their applicability to SMRs. The results are reported in [5]. It was concluded that EPR-related safety standards are technology neutral and apply fully to SMRs.

GSR Part 7 [4] establishes 26 requirements for the effective implementation of EPR arrangements. It defines the goals of emergency preparedness and emergency response and establishes general requirements, functional requirements and requirements for infrastructure.

The following IAEA Safety Standards provide Member States with recommendations on how to meet the requirements in GSR Part 7 [4]:

- IAEA Safety Standards Series No. GSG-2, Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency[7];
- IAEA Safety Standards Series No. GS-G-2.1, Arrangements for Preparedness for a Nuclear or Radiological Emergency [8];
- IAEA Safety Standards Series No. GSG-11, Arrangements for the Termination of a Nuclear or Radiological Emergency [9];
- IAEA Safety Standards Series No. GSG-14, Arrangements for Public Communication in Preparedness and Response for a Nuclear or Radiological Emergency [10].

GSG-2 [7] includes the definition of:

- Generic criteria on projected dose to implement response actions;
- Generic criteria on received dose to implement medical actions;
- Reference levels on residual dose as a basis to optimize, and to assess the effectiveness of, a protection strategy.

Those radiation protection concepts serve as basis for taking urgent and early response actions in emergency planning zones. Operational intervention levels, which are used for taking protective actions and other response actions based on environmental monitoring following a release of radioactive materials, are derived from the generic criteria.

Table 8 in GS-G-2.1 [8] provides suggestions for EPZ sizes for nuclear reactors, based on their nominal thermal power.

3.1.2. Graded approach and hazard assessment

Requirement 4 concerning hazard assessment in GSR Part 7 [4] states that:

> “**The government shall ensure that a hazard assessment is performed to provide a basis for a graded approach in preparedness and response for a nuclear or radiological emergency.**”

The application of a graded approach is intended to help design and implement arrangements that are commensurate with the level of hazard identified, as output from the hazard assessment. The hazard assessment needs to consider events that might require on-site actions to mitigate the release of radioactive materials and off-site response actions to protect the public and the environment, including:

- Events of very low probability and events not considered in the design, which include those triggered by a nuclear security event;
- Events involving a combination of a nuclear or radiological emergency with a conventional emergency that could affect wider areas and/or could impair capabilities to provide support to the response;
- Events affecting several facilities and/or modules concurrently, as well as interactions between them;
- Events at such facilities and/or modules located abroad.

In this connection, GSR Part 7 [4] defines five emergency preparedness categories (EPCs) (I–V), based on the nature and magnitude of the radiological consequences that emergencies might lead to. For a given facility, the relevant EPC is identified by the regulatory body based on the conclusions from the hazard assesment.

3.1.3. Emergency preparedness categories and associated protective actions

From the five EPCs defined in detailed in Table 1 of GSR Part 7 [4] (table notes omitted), EPCs I, II and III are applicable to nuclear facilities, including SMRs.

EPC I:

> “Facilities, such as nuclear power plants, for which on-site events (including those not considered in the design) are postulated that could give rise to severe deterministic effects off the site that would warrant precautionary urgent protective actions, urgent protective actions or early protective actions, and other response actions to achieve the goals of emergency response in accordance with international standards, or for which such events have occurred in similar facilities.” [4]

EPC II:

> “Facilities, such as some types of research reactor and nuclear reactors used to provide power for the propulsion of vessels (e.g. ships and submarines), for which on-site events are postulated that could give rise to doses to people off the site that would warrant urgent protective actions or early protective actions and other response actions to achieve the goals of emergency response in accordance with international standards, or for which such events have occurred in similar facilities. Category II (as opposed to category I) does not include facilities for which on-site events (including those not considered in the

design) are postulated that could give rise to severe deterministic effects off the site, or for which such events have occurred in similar facilities.” [4]

EPC III:

“Facilities, such as industrial irradiation facilities or some hospitals, for which on-site events are postulated that could warrant protective actions and other response actions on the site to achieve the goals of emergency response in accordance with international standards, or for which such events have occurred in similar facilities. Category III (as opposed to category II) does not include facilities for which events are postulated that could warrant urgent protective actions or early protective actions off the site, or for which such events have occurred in similar facilities.” [4]

Facilities in EPC I or EPC II, because of the potential occurrence of emergencies that might warrant taking preacutionary urgent or urgent protective actions, are required to have EPZs in place:

— A precautionary action zone (PAZ), defined as “an area around a facility for which emergency arrangements have been made to take urgent protective actions in the event of a nuclear or radiological emergency to avoid or to minimize severe deterministic effects off the site. Protective actions within this area are to be taken before or shortly after a release of radioactive material or an exposure, on the basis of prevailing conditions at the facility” [4], for facilities in EPC I; and
— An urgent protective action planning zone (UPZ), defined as “an area around a facility for which arrangements have been made to take urgent protective actions in the event of a nuclear or radiological emergency to avert doses off the site in accordance with international safety standards. Protective actions within this area are to be taken on the basis of environmental monitoring — or, as appropriate, prevailing conditions at the facility” [4], for facilities in EPCs I and II.

In addition, facilities in EPCs I and II, because of the potential occurrence of emergencies that might warrant taking early protective actions, are required to have emergency planning distances in place:

— An extended planning distance (EPD), defined as the “Area around a facility for which emergency arrangements are made to conduct monitoring following the declaration of a general emergency and to identify areas warranting emergency response actions to be taken off the site within a period following a significant radioactive release that would allow the risk of stochastic effects among members of the public to be effectively reduced” [4], for facilities in EPCs I and II; and
— An ingestion and commodities planning distance (ICPD), defined as the “Area around a facility for which emergency arrangements are made to take effective emergency response actions following the declaration of a general emergency in order to reduce the risk of stochastic effects among members of the public and to mitigate non-radiological consequences as a result of the distribution, sale and consumption of food, milk and drinking water and the use of commodities other than food that might have contamination from a significant radioactive release” [4], for facilities in EPCs I and II.

3.1.4. Fundamentals of the source term, and other parameters relevant for determining emergency planning zones and protective actions

The 2013 EPR Series publication, Actions to Protect the Public in an Emergency due to Severe Conditions at a Light Water Reactor (EPR-NPP-PPA) [11], provides technical guidance on how to size a PAZ and a UPZ for light water reactors (LWRs) with a nominal thermal power equal to, or higher than, 100 MW (th). Outlines from Appendix I in [11] can be applied to SMRs, even though additional considerations on the latest best practices, for example, in the area of modelling the atmospheric dispersion of radioactive materials using past numerical weather data, could be taken into account.

In the IAEANuclear Safety and Security Glossary [12], source term is defined as "The amount and isotopic composition of *radioactive material* released (or postulated to be released) from a *facility*."

The hazard assessment might lead to building several accident scenarios or sequences in order to cover the wide spectrum of hazards identified, each scenario or sequence having a corresponding source term. For the atmospheric dispersion of a given source term, additional release characteristics, such as the release height, the release start and duration, and local weather conditions or monitored numerical weather data, need to be considered. Regarding the dose projection associated to this dispersion of radionuclides from a release, due consideration needs to be given to the exposure pathways, as well as the group(s) of people and dose conversion factors. Lastly, the generic criteria in place as per national regulations enable conclusions on technical suggestions for the EPZ size.

3.2. PRESENT APPROACHES TO DETERMINE EPZs IN MEMBER STATES AND THEIR APPLICABILITY TO SMRs

This section aims to provide an overview of the regulatory framework in place and/or under development for the deployment of SMRs in Member States. Most EPZ regulatory frameworks fall into one of the following two categories: a framework where EPZ sizes are uniformally nationally prescribed and a framework where EPZ sizes are based on a case by case off-site consequence assessment and rely on the decision by the responsible authority. The research carried out by the participating institutes is described in Section 3.3. The research does not necessarily follow the relevant national regulatory framework that is described below.

3.2.1. Canada

In Canada, the respective roles of the various levels of government in nuclear emergency management are derived from legislated responsibilities. Through the authority of the Nuclear Safety and Control Act (NSCA), the Canadian Nuclear Safety Commission (CNSC) regulates the Canadian nuclear industry to prevent unreasonable risk to the environment, the health and safety of persons, and national security. Health Canada, as the lead department under the Federal Nuclear Emergency Plan, provides guidelines for intervention following a nuclear emergency in Canada or affecting Canadians in the generic criteria and operational intervention for nuclear emergency planning and response [13]. These guidelines are key references for provincial governments when preparing provincial nuclear emergency plans, as well as for other responsible agencies and applicants for licences for activities regulated under the NSCA.

Provincial and territorial governments bear the primary responsibility for protecting public health and safety, property and the environment within their borders and are ultimately responsible for the off-site response and the implementation of protective actions and measures. It is understood and practiced in Canada that the off-site authorities determine the size of the EPZ. However, regardless of the chosen size of EPZ, the requirements for emergency planning will be based on the facility's emergency planning basis and the potential off-site impacts.

In Canada, the EPZ consists of Automatic Action Zone, Detailed Planning Zone, Contingency Planning Zone and Ingestion Planning Zone. These zones correspond with the IAEA's precautionary action zone, urgent protective action planning zone, extended planning distance, and ingestion and commodities planning distance respectively.

3.2.1.1. Regulatory approach to EPZ

The Canadian regulatory approach is built on a long-established foundation of risk-informed regulations. Regulatory tools and decision-making processes are structured to enable a licence applicant for a nuclear facility to propose alternative ways to meet regulatory expectations. Proposals must demonstrate, with suitable information, that they are equivalent to or exceed regulatory requirements. Current requirements and guidance for nuclear facilities are generally articulated to be technology neutral and permit the use of a graded approach that enables applicants to establish the stringency of design measures, safety analyses and provisions for conduct of their activities commensurate with the level of risk posed by the facility. For example, Regulatory Document (REGDOC) 2.10.1, Emergency Preparedness and Response [14] has been recently updated to reflect neutral technology (the final version of this update is still pending publication).

Although the decision on EPZ sizing is within provincial/territorial jurisdiction, CNSC's regulatory framework includes requirements for the different licensing phases to support EPZ sizing determination in which applicant and/or licensee must meet.

For Licence to Prepare Site, the applicant is required to conduct a site evaluation and consider emergency planning, as well as hazard analysis, assessment of accidents, and exclusion zone and EPZ information. At this licensing phase, the CNSC is also looking for documented agreements from the off-site authorities that their plans can accommodate the life cycle of the proposed project, including their communication strategies and processes.

For licence to construct phase, the applicant would need to submit a Preliminary Safety Analysis Report (PSAR). The PSAR would include the identification of postulated initiating events (PIEs) using a systematic methodology. The proposed EPZ sizing is to be based on a spectrum of accidents from the PSAR.

For the licence to operate phase, the applicant must demonstrate the full implementation of all regulatory requirements based on the Final Safety Analysis. This includes the finalization of the planning basis for the facility's Emergency Programme. The applicant is expected to submit a comprehensive detailed site-specific Emergency Programme with all respective agreements with off-site organizations in place.

3.2.1.2. EPZ size determination process

Applicants/licensees are required to provide regional and provincial off-site authorities with the necessary information to allow for effective emergency planning policies and help the province establish the appropriate EPZ around the nuclear facility. This information must be kept up to date on a periodic basis and include the limiting credible accidents and their associated source terms in accordance with REGDOC 2.10.1[14].

In addition to the mechanistic source terms determination from the credible accidents identified within the planning basis, provincial/territorial authorities would also consider external factors such as social impacts, geography of the location, and the population demographics in determining the EPZ and sizing around the nuclear facility.

Even if it has been determined that EPZ is not required based on the planning basis which demonstrates that there would be no off-site radiological consequences requiring the implementation of protective actions, an EPD should still be considered to address the unforeseen events. This is a site-specific distance around a nuclear facility within which arrangements are made, such as to conduct early monitoring of deposition to confirm that off-site protective actions are not warranted or would be required following a release from the facility.

3.2.2. China

On-site emergency plans developed by NPPs and off-site emergency plans formulated by provincial governments include information regarding EPZs for NPPs. According to the Law of the People's Republic of China on Nuclear Safety, and Regulations on Response to Nuclear Emergencies at Nuclear Power Plants (HAF 002-2011), the on-site emergency plans for nuclear accidents are reviewed by the national nuclear safety regulator, i.e. National Nuclear Safety Administration (NNSA), while off-site emergency plans for nuclear accidents are reviewed by the National Nuclear Emergency Coordination Committee.

According to the National Emergency Response Plan for Nuclear Accidents, the size of EPZs is proposed by the NPP and, after agreement from the NNSA and review by the provincial nuclear emergency committee, it is submitted for approval to the National Nuclear Emergency Coordination Committee. Off-site protective actions such as sheltering, intake of stable iodine, control of pathways, control of food and water, evacuation, relocation and decontamination of affected areas, are implemented by departments designated by the provincial government.

Emergency Preparedness and Response for Nuclear-Powered Facility Operators (HAD 002/01-2019), Criteria Emergency Planning and Preparedness for Nuclear Power Plant — Part 1: The Dividing of Emergency Planning Zone (GB/T 17680.1-2008) and the National Emergency Response Plan for Nuclear Accidents, provide the size and technical requirements of EPZs for large NPPs. After the 2011 Fukushima Daiichi NPP accident, for large reactors in China the plume exposure pathway EPZs were determined according to the national upper limits recommended in the national standard, with an inner zone of 5 km and an outer zone of 10 km, and the ingestion pathway EPZ was typically set at 50 km.

Principles for Safety Review of Small Pressurized Water Reactor Nuclear Power Plants (Trial), issued by the National Nuclear Safety Administration in 2016 (referred to as the Review Principles), states the need to further enhance the safety of small pressurized water reactor (PWR) NPPs to simplify the off-site emergency response. China's view is that PWR NPPs are likely to provide the public with a higher level of protection than large pressurized water reactor NPPs.

In October 2017, the National Nuclear Accident Emergency Office of China issued the Emergency Instruction Opinions on Inland Small Pressurized Water Reactor Nuclear (Trial), which stipulates the principles to be followed in a nuclear emergency, methods for defining the emergency planning zone and optimization of emergency preparedness.

3.2.2.1. Challenges in developing a regulatory framework for SMR EPZ

When developing the regulatory framework for SMR emergency planning zones, China faces the following challenges:

- Improving the development of regulations for SMRs. China has relatively few guidance publications related to emergency management of SMRs, and there is a significant gap in the systematic and comprehensive nature of SMR emergency regulations compared to those for large reactors. It is necessary to further develop regulations for SMRs.
- Investigating the suitability of the EPZ regulatory requirements and calculation methods for large reactors that apply to SMRs. In general, SMRs have inherent safety characteristics, such as lower core power, lower radioactive inventory and passive safety design, which make the deterministic method used in defining emergency planning zones for large reactors unsuitable for SMRs. Thus, further research will be conducted on the calculation methods for and technical details of emergency planning zones for SMRs.

3.2.2.2. Proposed modifications for SMR regulation

Currently, several institutes have been conducting design work for various types of SMRs in China, with construction already under way for some SMRs. The nuclear safety regulator plans to propose a management requirement for a performance-based approach for SMRs that is different from that for large reactors. This is based on the conclusions of a safety analysis and a cost–benefit analysis, considering the safety characteristics and application specific aspects of SMRs. In terms of the calculation method for the EPZ size, a combination of probabilistic risk assessment and deterministic dose assessment will be considered to decide this for SMRs in a more rational and scientific manner.

3.2.3. Finland

A revised EPZ size regulation in Finland was updated in February 2024 to be based on analysis of the highest possible doses received by the population in different geographical ranges in case of an accident at a nuclear facility. Prior to this update, the regulation defined values of 5 km (PAZ) and 20 km (UPZ) for NPPs. NPPs that were in operation prior to 2024 might continue to apply these distances without new analysis.

The current regulation (STUK Y/2/2024 [15]) specifies that the licensee has to propose the sizes for PAZ and UPZ for approval to STUK and has to show that the following doses would not be exceeded outside the defined zones in case of accident:

— 1 Sv in 10 hours outside PAZ
— 10 mSv in 48 hours UPZ; however UPZ is at maximum 20 km even if this value would be exceded outside it.

Finland’s national requirements further state:

— Land use restrictions in the 5 km PAZ:
  - No schools, hospitals, care facilities, shops, or significant places of employment or accommodation (except the NPP); no densely populated areas;
  - No socially significant functions that could be affected by an NPP accident;
  - Number of permanent inhabitants, recreational housing and activities limited (to approximately 200) to allow effective evacuation;
  - Permanent and leisure time population must not increase substantially during NPP construction and operation.
— Reference to licensee duties within PAZ;
— The UPZ must have a detailed external rescue plan for the protection of the public.

Limits for radiation exposure of members of the public (population) are given in Section 22b of the Nuclear Energy Decree 161/1988, amendment 1001/2017 [16]:

> “The release of radioactive substances arising from a severe accident shall not necessitate large scale protective measures for the public nor any long-term restrictions on the use of extensive areas of land and water.
>
> In order to restrict long-term effects the limit for the atmospheric release of cesium-137 is 100 terabecquerel (TBq). The possibility of exceeding the set limit shall be extremely small.
>
> The possibility of a release in the early stages of the accident requiring measures to protect the public shall be extremely small.”

The STUK emergency response guidelines can be found in [17]. The Nuclear Energy Act and all related decrees and regulations are under full revision under the Ministry of Economic Affairs and Employment. The legislation is being revised to rely on a more technology neutral basis and to be in line with changes in technologies, other regulation, and other changes in Finland’s society. During the revision, it is not expected that the basis for determining the EPZ sizes would remain the same as in STUK Y/2/2024 [15]. However, some requirements or guidance might be added to specify in more detail what assumptions must or might be used for the accidents that are used as basis for the analysis and how the analysis should be conducted. The limits for radiation exposure of the members of public will be reviewed during the revision and might be changed to be more applicable to different types and sizes of facilities. The expected time for entry into force of full revision is early 2028.

3.2.4. Indonesia

In determining the EPZ for SMRs, the Nuclear Energy Regulatory Agency of Indonesia (BAPETEN) uses Chairman Regulation Number 1/2010 on the Emergency Preparedness and Response for Nuclear Emergency. This regulation provides the regulatory provisions for licensee(s) to prepare for the nuclear emergency preparedness and response programmes required to obtain a construction permit.

The emergency preparedness and response programme includes radiological hazard assessment that covers EPZ calculation. EPZ calculation is one of the essential assessments to estimate how far and how wide the radiological release from a nuclear reactor will be within a certain kilometre radius. The radiological release, in kilometre radius, is then compared to the surrounding area of the nuclear installation to identify radiological hazards for nearby residences, the environment, land use and spatial planning. The potential radiological hazards are then assessed to formulate mitigative action required to control and mitigate the radiological consequences for nearby residences, the environment, land use and spatial planning.

In calculating the EPZ, BAPETEN does not restrict licensee(s) on the approach and methodology to use as long as the calculation satisfies the reference value stipulated in Chairman's Regulation Number 1/2010, Appendix 2. BAPETEN in this case encourages the licensee to demonstrate that technological improvement and/or design innovation enhance the reactor safety performance and reduce the area of the EPZ.

A risk informed and performance-based approach is used by BAPETEN to review and evaluate the EPZ calculation submitted by the licensee in the emergency preparedness and response programme. In this review, BAPETEN conducts verification of the calculation and discusses the method and approach used by the licensee. BAPETEN then will issue approval for the EPZ based on the discussion and verification.

From BAPETEN Chairman Regulation Number 1/2010 Appendix 2, there are two categories of EPZ area for nuclear power reactors. The first is for conventional large reactors with a thermal output of more than 1000 MW(th) and small to medium size reactors with a thermal output of 100–1000 MW(th). For conventional large power reactors, the minimum EPZ area is a 25 km radius, whereas for small and medium size reactors the EPZ area falls within the range of a 5–25 km radius. This EPZ area radius value act as a reference value to evaluate the adequacy of the licensee's calculation.

For small reactors such as SMRs, the value varies from 5–25 km. This range does not reflect the linearity between SMR thermal output and the EPZ area. However, this range serves as a reference value to determine the EPZ area. Hence, the EPZ for SMRs will be no less than 5 km and no more than 25 km. Using this reference level, a SMR licensee will demonstrate technological improvement and design innovation to achieve an EPZ that is as small as realistically possible.

In addition to technological improvement and design innovation, site selection also plays a crucial role in achieving optimal EPZs for SMRs. A site's meteorological data have to be considered in calculating the EPZ for an SMR. Thus, the calculation will make sure that if there is an early radiological release or a large release from the plant, the hazard can be contained within the EPZ radius. Therefore, the EPZ value will vary from one reactor to another based on reactor type, power output and the site's meteorological characteristics.

3.2.5. Republic of Korea

In the Republic of Korea, the Nuclear Safety and Security Commission (NSSC) first issued emergency preparedness guidelines in 1980 after the accident at the Three Mile Island NPP in the United States of America. The EPZ size for NPPs is specified to be 8–10 km from the site and arrangements for taking protective actions are required to be implemented in this area. After the accident at the Fukushima NPP in 2011 in Japan, the NSSC revised the guidelines, with the EPZ now being divided into the PAZ and UPZ following the IAEA Safety Standards. Currently, for a commercial NPP, the PAZ is specified to be a 3–5 km zone and the UPZ is specified to be a 20–30 km zone. For a research reactor with HANARO rated power of 30 MW(th) in the Republic of Korea, for example, the EPZ was set to 800 m initially but revised to 1.5 km after the Fukushima accident, based on the EPZ sizes suggested in GSG-11 [9]. Some protective actions specified by the NSSC, together with their dose criteria (including exposure pathway and dose accumulation time period), are as follows:

- Sheltering: 10 mSv of avertable dose over a maximum of 2 days;
- Evacuation: 50 mSv of avertable dose over a maximum of 1 week;
- Iodine prophylaxis: 100 mGy of avertable committed absorbed dose to the thyroid;
- Relocation: 30 mSv over a month, or a lifetime dose exceeding 1 Sv.

3.2.6. United Kingdom

In the United Kingdom the arrangements for defining emergency planning zones are covered by the Radiation (Emergency Preparedness and Public Information) Regulations 2019 (REPPIR). These regulations transpose the requirements of the Euratom Basic Safety Standards Directive into United Kingdom law. The regulations are intended to be technology neutral and apply to all types of nuclear facilities currently operating in the United Kingdom; these include NPPs, fuel cycle facilities, waste facilities and defence related activities. The current intent is that these regulations would also apply to future SMRs.

The REPPIR set introduces a graded approach to off-site emergency planning. For higher hazard facilities, this comprises some combination of 'detailed' planning for the most likely accidents and 'outline' planning to provide mitigation against more severe very low probability events, for example those potentially not considered in the design. For this purpose, the requirement is to determine, based on hazard analysis, a detailed emergency planning zone (DEPZ) and a larger outline planning zone. Detailed planning requires the local authorities, supported by emergency responders and government agencies, to prepare a site specific emergency plan. The plan should specify the urgent protective actions (evacuation, sheltering, stable iodine) and the geographical extent/timescales for these.

The central aim of outline planning is to support the decision making of emergency responders in the event that detailed or any generic emergency arrangements are not sufficient. Outline planning, which is likely to be required over a wider area, is about identifying what protective actions might be needed at a strategic level, where those capabilities could be obtained from and the anticipated time frame over which they will become available, rather than having them in place ready to mobilize without delay.

The size of the DEPZ is determined by the operator using a risk informed approach defined in REPPIR. Using existing deterministic and probabilistic safety analysis where appropriate, this requires the operator to identify and quantify the full range of events that could give rise to a radiation emergency. For facilities such as NPPs, detailed plans are required for all events with a likelihood of occurrence of ~1 in 100 000 years or greater. The expectation is that any events just beyond this limit should also be considered where there is the potential for a 'cliff edge' increase in consequences.

For the bounding event, the extent of the DEPZ is determined by the operator using a prescribed approach covering pathways, targets, exposure assumptions and dose criteria. Specific calculation models are not prescribed, although models considered to reflect good practice are identified and their use is encouraged. The extent of urgent protective actions is based on the United Kingdom's emergency reference levels for 'averted dose'. Application of the emergency reference levels allows a balance to be made between the benefits of averting dose through the protective action and the potential disbenefits that would be associated with implementing immediate protective actions across too wide an area. The expectation is that dispersion modelling should take account of pessimistic consequences due to unfavourable weather conditions via the use of the 95th percentile of consequences based on weather variability.

As an extension of the detailed plan, outline planning is required for lower frequency events with more severe consequences. REPPIR specifies the distances for the outline planning zones to be used for defined categories of facilities. The specified distances for each category have been derived based on modelling of severe events considered to be bounding for the purposes of practical emergency planning. A single distance is specified covering all NPPs, which would include future SMRs, although provision is made in the regulations for operators to agree (if justified) an alternative outline planning zones distance with the Government/regulator.

3.2.7. United States of America

In the United States of America, the Nuclear Regulatory Commission (NRC) emergency preparedness regulations for large LWRs include standards for both on-site and off-site emergency response plans. The regulations in Title 10 of the Code of Federal Regulations (CFR) Part 50 [18], specifically § 50.47 and § 50.54, prescribe how the NRC makes licensing decisions using findings of reasonable assurance that adequate protective measures can and will be taken to protect public health and safety in the event of a radiological emergency. The NRC will not issue an initial operating licence unless a finding of reasonable assurance can be made. A reasonable assurance finding is based on, first, the NRC's assessment of the adequacy of the applicant's or licensee's on-site emergency plan and whether there is reasonable assurance the plan can be implemented and, second, for applicants and licensees with an EPZ beyond the site boundary, the NRC's review of the Federal Emergency

Management Agency findings and determinations as to whether State and local emergency plans are adequate, and whether there is reasonable assurance that they can be implemented.

An EPZ bounds the area surrounding a facility within which detailed preplanning is needed to implement predetermined, prompt protective actions. EPZs are used to simplify protective action decision making when such decisions might be time constrained or complex. EPZs are scalable in size. However, the EPZ size does not change the requirements for emergency planning; it only sets physical bounds on the planning. Detailed planning within the EPZ provides a basis for expansion of the response efforts beyond the EPZ, should it become necessary. Risk insights from design basis accidents and probabilistic risk assessments led to a generic 10 mile plume exposure pathway EPZ for large LWRs. NRC regulations provide for a case by case determination of the size of the plume exposure pathway EPZ for gas cooled reactors and for reactors with an authorized power level <250 MW(th). The same risk informed methodology that informed the EPZ size for LWRs can be applied to other facility types. The methodology includes an analysis of dose versus distance, comparing projected dose from a spectrum of accidents to the dose criteria for recommending protective actions. The EPZ size is also informed by considering the likelihood of exceeding life threatening doses for more severe beyond design basis accidents.

Staff in the NRC have developed a draft performance based emergency planning framework for SMRs and other new technologies as an alternative to the current framework developed for LWRs [19]. The performance-based emergency planning framework goals are:

- To provide reasonable assurance that adequate protective measures can and will be implemented by an SMR or other new technologies licensee;
- To promote regulatory stability, predictability and clarity;
- To reduce the need for exemptions from emergency planning requirements;
- To recognize advances in design and technology;
- To credit safety enhancements in evolutionary and passive systems; and
- To credit the potential benefits of smaller sized reactors and non-light water reactors associated with postulated accidents, including slower transient response times and relatively small and slow releases of fission products.

The draft performance-based requirements address the most risk significant aspects of emergency planning (i.e. classification, notification, accident assessment, protective actions) and their supporting planning elements. Regulatory compliance will be verified using the results of licensee conducted drills and exercises and inspection of a set of quantified emergency response function performance objectives.

The performance-based framework also includes a provision for scalable EPZs. Under the performance-based framework, the EPZ is the area within which:

- Public dose is projected to exceed 10 mSv total effective dose equivalent (TEDE) over 96 h from the release of radioactive materials considering accident likelihood and source term, timing of the accident sequence and meteorology; and
- Predetermined, prompt protective measures are warranted.

The first criterion facilitates a dose versus distance analysis for determination of the EPZ size, if an EPZ is needed. Guidance for informing the EPZ size is based on the methodology developed in Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants (NUREG-0396) [20]. The proposed methodology includes guidance for accident and source term selection, meteorological input, atmospheric transport modelling, exposure parameters and dose criteria. The proposed guidance also addresses the use of uncertainty analyses and considerations for accident likelihood and timing as it relates to the EPZ size determination. For example, it might be possible to exclude accident sequences from EPZ size determinations due to an extremely low likelihood of occurrence or an extended amount of time available for initiating protective actions associated with the accident sequence. However, even if a particular accident sequence does not inform the EPZ size, the accident characteristics — particularly the timing and materials released — would still inform other EP functional capabilities and ensure some capability for response for even the most severe accidents.

The second criterion is a functional requirement for the EPZ. Because SMRs and other new technologies are expected to have different accident timing characteristics from large LWRs, an assessment of the functional criteria will determine whether an EPZ is needed. For example, prompt protective actions might not be required based on the time available to initiate ad hoc protective measures. As such, it might be possible to demonstrate through a functional analysis that no EPZ is needed to support the planning. However, a site boundary EPZ, or no EPZ does not imply that protective actions are precluded off-site. To meet the second criterion an applicant has to demonstrate that sufficient time and capability exists (e.g. accident assessment, dose projections, communications, protective action recommendations) to initiate protective measures off-site without the need for detailed preplanning. The second criterion places the emphasis on capabilities over prescriptive distances.

The emphasis on capabilities in a performance-based framework is also reflected in the planning for the ingestion pathway. In lieu of an ingestion pathway EPZ, the performance-based framework would require applicants and licensees to describe the resources and capabilities to prevent contaminated food and water from entering the ingestion pathway in their emergency plans. The performance-based framework focuses on the functional preparedness aspects for exposure pathways. Doing so ensures that the EPZ concept is properly applied and scalable to support planning and response.

In cases where a plume exposure pathway EPZ does not extend beyond the site boundary, the NRC has confidence in the ability of off-site response organizations to implement appropriate protective actions as necessary using comprehensive ‘all hazards’ emergency planning. Protective actions taken by off-site response organizations for a radiological emergency (e.g. evacuation, sheltering) are the same actions as would be taken for other emergency, such as wildfires or hurricanes. The NRC’s confidence is expressed in the NRC’s regulations in § 50.47(c)(1)(iii) and further strengthened by the NRC’s recognition and support of national level efforts (e.g. National Incident Management System [21], National Preparedness Goal [22], Core Capabilities [23], National Preparedness System [24], National Planning Frameworks [25]), in which the NRC participates, to improve the state of emergency planning at all levels of Government and within the whole community [26].

3.2.8. European Commission, Joint Research Centre

There are two fundamental legal acts at European Union (EU) level concerning emergency planning for nuclear facilities: the Basic Safety Standards [27]and the European Nuclear Safety Directive [28].

Both legal acts are directives, namely legislative acts that set out a goal for all EU countries to achieve. However, it is up to the individual countries to devise their own laws on how to reach these goals. This means that EU directives needs to be transposed at national level.

Reference [27] states that:

> "With regard to the management of emergency exposure situations, the current approach based on intervention levels should be replaced by a more comprehensive system comprising an assessment of potential emergency exposure situations, an overall emergency management system, emergency response plans, and pre-planned strategies for the management of each postulated event.
>
> "The introduction of reference levels in emergency and existing exposure situations allows for the protection of the individual as well as consideration of other societal criteria in the same way as dose limits and dose constraints for planned exposure situations.
>
> …….
>
> "Member States shall ensure that reference levels are established for emergency and existing exposure situations. Optimisation of protection shall give priority to exposures above the reference level and shall continue to be implemented below the reference level.
>
> "The values chosen for reference levels shall depend upon the type of exposure situation. The choices of reference levels shall take into account both radiological protection requirements and societal criteria. For public exposure the establishment of reference levels shall take into account the range of reference levels set out in Annex I.
>
> …….
>
> "Without prejudice to reference levels set for equivalent doses, reference levels expressed in effective doses shall be set in the range of 1 to 20 mSv per year for existing exposure situations and 20 to 100 mSv (acute or annual) for emergency exposure situations."

Despite no further technical dispositions regarding EPZs applying at the EU level — for example, there are no regulations on specific EPZ distances — the EU Nuclear Safety Directive [28]requires that:

"[N]uclear installations are designed, sited, constructed, commissioned, operated and decommissioned with the objective of preventing accidents and, should an accident occur, mitigating its consequences and avoiding:

(a) early radioactive releases that would require off-site emergency measures but with insufficient time to implement them;
(b) large radioactive releases that would require protective measures that could not be limited in area or time."

3.3. OVERVIEW OF RESEARCH CONDUCTED BY EACH PARTICIPATING INSTITUTE

3.3.1. National Atomic Energy Commission, Argentina

The National Atomic Energy Commission (CNEA) has used a figure of merit called the individual radiological risk (IRR) and calculated the iso-risk curves at the NPP surroundings. This figure of merit is an extension of the IRR used by the Regulatory Authority for NPP licencing process.

Several parameters in terms of dose and risk were compared in order to identify a figure of merit for the EPZ boundaries assessment. IRR is equal to the probability of exposure for members of the public (which depends on e.g. the type of exposure, plant characteristics, meteorological conditions, topography) times the probability of fatality due to the exposure.

3.3.2. Canadian Nuclear Laboratories, Canada

Canadian Nuclear Laboratories (CNL) performed research in four areas:

— Accident phenomenology;
— Source term calculation;
— Atmospheric dispersion and deposition;
— Dose projection.

3.3.2.1. Accident phenomenology

The CNL study focused on phenomena expected to occur in postulated reactor accidents to improve source term predictions for those accidents. These efforts focused on two areas: air ingress in HTGRs, and fission product aerosol behaviours in integral pressurized water reactors (iPWRs).

A phenomena identification and ranking table exercise was previously performed at CNL for a postulated severe accident in a generic HTGR. This study concluded that the amount and timing of air ingress following a primary circuit break was critical for predicting the accident source term because of the expected core graphite oxidation. The phenomena identification and ranking table study also concluded that predictive capability for air ingress was lacking, along with experimental data suitable for validating model predictions. As part of this research, a new test apparatus was conceived with the intent of performing new experiments inside CNL's Large Scale Containment Facility. A conceptual design of the High Temperature Air Ingress Test Facility (HTAIF) was completed, but problems with the design identified during detailed technical review, and the excessive cost of constructing the HTAIF as designed, halted progress. The research effort was redirected towards component proof of concept tests, including investigation of a non-invasive helium–air concentration measurement technique, and experimental validation of novel electrical heaters for the HTAIF.

On the topic of iPWR source term investigations, the conditions inside the small containment vessels, specifically the large surface area to volume ratio and the significant condensation heat transfer on the containment vessel wall, are drastically different from the conditions inside contemporary large water cooled reactors with concrete containment structures. Consequently, fission product aerosol removal phenomena such as diffusiophoresis, thermophoresis and turbulent deposition are expected to be much more impactful towards accident source terms than in past analyses. The Strong Condensation Containment Apparatus (SCCA) was constructed inside CNL’s Large Scale Containment Facility to study containment thermohydraulic and aerosol transport phenomena under these conditions, as well as provide data that can be used to validate predictive models. The first experimental campaign was completed in early 2021, wherein transport and deposition of fission product aerosol simulants were effectively measured under the desired conditions. These results have been published in the open literature [29].

3.3.2.2. Source term calculation

A pilot study on the radiation dose consequences of postulated limited accidents in several SMRs was performed by CNL to assess the possible implications on EPZ sizes. The reactor technologies considered in this pilot study included a HTGR, an iPWR, a MSR and a lead cooled fast reactor (LFR). An extensive literature review was performed to identify release fractions of radionuclides in the postulated accidents. The release fractions were then applied to initial core radionuclide inventories calculated by CNL in past studies to produce source terms for these limiting accidents. The source terms (as part of the pilot study) have been published in the open literature [30].

(a) Atmospheric dispersion and deposition

Reliable predictions of radionuclide dispersion and dose consequence in the environment surrounding a NPP are important for determining the required extent of the EPZ. If SMRs are to have small EPZs, then relatively short ranged dispersion and dose assessments might be necessary. Resolving the so called ‘near field’ distances of a few hundred metres is challenging for common dispersion models that operate on the scale of tens of kilometres for assessment of contemporary NPP accidents. CNL has, therefore, performed development of a fine resolution atmospheric dispersion modelling platform.

The platform has four different dispersion modelling options (Gaussian plume and puff, Lagrangian particle, polydisperse Gaussian moment (PGM)), and integrates the United States Environmental Protection Agency’s CALMET module for reconstructing wind and temperature fields in the atmosphere. As such, it has the capability to model dispersion over complex terrain and can incorporate dynamic meteorology data. Advanced metrological measurement techniques such as sonic detection and ranging (SODAR) and drone mounted anemometers were subsequently investigated because they can provide sufficient data density to reconstruct more complex local wind fields. The platform also incorporates three-dimensional dose calculation methods, which is essential when considering near field dose in the vicinity of a source because typical dose conversion factors for ground shine and cloud shine rely on assumptions of homogeneity in the source distribution around a receptor. The dose calculation is performed with ninth order Gaussian quadrature integration on continuous fields (Gaussian plume and puff dispersion), or summation on discrete solutions (Lagrangian particle and PGM). Whereas the computational complexity of such schemes would have precluded application to emergency preparedness and response in the past, significant speedup has been achieved through advanced techniques and parallelization on modern computational hardware.

Predictions from the modelling platform will be benchmarked against historical ^{41}Ar plume measurements on the Chalk River Laboratory site resulting from the operation of the National Research Universal (NRU) reactor.

(b) Dose projection

As part of the pilot study on postulated limiting accidents in several SMR technologies, the literature derived source terms for HTGR, iPWR, MSR and LFR accidents were provided as input to the Atmospheric Dispersion and Dose Analysis Method (ADDAM) computer code to assess potential dose consequences to the public. The ADDAM code is part of the industry standard toolset for nuclear safety analysis in Canada and is fully consistent with the Canadian standard CSA N288.2 Guidelines for Calculating the Radiological Consequences to the Public of a Release of Airborne Radioactive Material for Nuclear Reactor Accidents [29]. Measured meteorological conditions on the Chalk River Laboratory site were used as input for the dispersion calculations.

The primary output of interest was the distance from the reactor where specific effective doses, corresponding to different protective action and regulatory limits, were reached for each of the model source terms and release conditions. As a benchmark, these were compared against previous calculations of the dose consequence from an unmitigated station blackout at a Canada deuterium–uranium (CANDU) plant on the same site (note that this was for exploratory purposes only because no such reactor exists at Chalk River Laboratory). The distances at which the doses were reached were much smaller for the SMRs, providing evidence that EPZs could be much smaller for these reactors than for large NPPs. The results of this study have been published in the open literature [30].

3.3.3. China Nuclear Power Engineering Corporation, China

China Nuclear Power Engineering Corporation (CNPE) conducted its research considering a 125 MW(e) PWR type SMR called ACP100, using a probabilistic approach. For that purpose, and in the absence of regulatory requirement for a cut-off frequency for selecting accident scenarios from probabilistic safety assessments (PSAs), CNPE used a 10^{-7} to 10^{-8} per reactor-year ‘temporary’ cut-off frequency.

The scenarios considered comprised DBA scenarios with higher radiological consequences for the purpose of EPZ sizing and severe accident scenarios. The dose criteria considered for assessing off-site radiological consequences were those currently in place in China: 10 mSv effective dose for sheltering, 50 mSv effective dose for evacuation and 100 mGy at the thyroid for the ingestion of iodine thyroid blocking agent.

As an initial position prior to conducting the research, CNPE anticipated that the EPZ size would be <3 km. Based on the accident scenarios selected and the dose criteria applicable, CNPE ran dose projection simulations and obtained EPZ size estimations, considering realistic weather conditions, conservative weather conditions, stochastic health effects and deterministic effects of a release.

3.3.4. Shanghai Nuclear Engineering Research and Design Institute, China

Research contributions from the Shanghai Nuclear Engineering Research and Design Institute (SNERDI) used the CAP200 reactor as a basis for investigating SMR EPZ requirements and for performing an EPZ case study. The CAP200 is a compact two loop small reactor with passive safety systems. The specific research performed by SNERDI on EPZ methodology and application to the CAP200 reactor is described below.

3.3.4.1. Accident cut-off probability and source term

The EPZ size needs to be based on the comprehensive characteristics of the reactor and accident analysis. Accidents considered for SMRs should include design basis accidents and severe accidents. Probabilistic risk assessment is generally used for severe accidents. The source term of the accident needs to be determined in accordance with relevant national regulations, and the analytical methods and procedures have to be approved by the national nuclear safety regulatory organization.

The draft publication, Notice of the National Nuclear Accident Emergency Office on Printing and Distributing the Guiding Opinions on Land-based Small PWR Nuclear Emergency Response, states that design basis accidents and severe accidents should be considered. The severe accident spectrum might temporarily take 10^{-7} to 10^{-8} per reactor-year as a cut-off value for determining the EPZ. Based on this cut-off, a core damage sequence for the CAP200 could not be screened out, and the containment could be assumed to be intact.

3.3.4.2. Atmospheric diffusion model

The MACCS program, developed by the United States NRC for severe accident consequence evaluation, was the main calculation and analysis tool. MACCS consists of three modules: ATMOS, EARLY and CHRONC. ATMOS estimates plume transport and dispersion using a Gaussian model. EARLY calculates dose, acute health effects and dose mitigation through urgent protective actions. CHRONC mainly focuses on long term dose, chronic health effects, dose mitigation through early response actions and the associated economic costs.

The dispersion parameters needed to be highly optimized for the CAP200 accident evaluation in order to predict doses lower than 10 mSv beyond the site boundary. For this reason, a custom computational fluid dynamics (CFD) model, MSDA-3D, was developed. The CFD model was verified with a series of wind tunnel experiments. Doses predicted by the CFD model were less than half of those predicted by MACCS.

3.3.4.3. Identification of acceptance criteria

China’s draft Guidelines for the Emergency Framework for SMR (National Nuclear Responsibility Office No. 29 (2017) states that the EPZ for a SMR should follow the overall principles and general methods described in Nuclear Power Plant Emergency Planning and Preparation Guidelines Part 1: Division of Emergency Planning Zone (GT/T 17680.1-2008). Considering all factors comprehensively, the proposed size of a SMR EPZ should not be larger than 3 km. The specific size should be based on systematic argumentation and scientific calculations and determined according to prescribed procedures. A single enveloping EPZ should be considered for multiple SMR modules.

Requirements for the exclusion area boundary (EAB), low population zone (LPZ) and EPZ that were identified for contemporary large nuclear power plants and for SMRs in China are summarized in Table 1.

TABLE 1. REQUIREMENTS FOR THE EAB, LPZ AND EPZ IDENTIFIED FOR CONTEMPORARY LARGE NUCLEAR POWER PLANTS AND SMRs IN CHINA

	Nuclear power plant requirements		**SMR requirements**
EAB	Postulated siting accident: EAB dose should not exceed 0.25 Sv within 2 h. LPZ dose should not exceed 0.25 Sv within 30 d. Collective dose should not exceed 2×10^4 person Sv within 80 km	≥500 m ≥5000 m	Infrequent accidents: site boundary effective dose limit is 5 mSv within 30 d Limiting accidents: site boundary effective dose limit is 10 mSv within 30 d
LPZ (GB6249)	Infrequent accidents: LPZ effective dose limit is 5 mSv, thyroid dose limit is 50 mSv Limiting accidents: LPZ effective dose limit is 0.1 Sv, thyroid dose limit is 1 Sv Sheltering <2 d, 10 mSv	3–5 km (internal exposure)	Severe accidents: site boundary effective dose limit is 10 mSv within 30 d
EPZ (plume) (EPZ_ (GB/T 17680.1)	Evacuation, <7d, 50 mSv Stable iodine intake, 100 mSv	7–10 km (external exposure)	

3.3.4.4. CAP200 EPZ size case studies

One coastal site (Shidaowan, Shangdong Province) and two inland sites (Taohuajiang, Hunan province and Pengze, Jiangxi Province) were chosen for case studies to illustrate the calculation process for the CAP200 SMR EPZ.

The calculation results showed that the probabilities of the dose exceeding 10 mSv at 500 m from a site were 0.54%, 0.76% and 0.06%, respectively, for the three sites. These were much smaller than the 30% exceedance probability at 10 miles suggested in Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants (NUREG-0396) [18]. At a distance of 500 m, the average peak individual dose for the Shidaowan, Taohuajiang and Pengze sites was 4.74 mSv, 5.33 mSv and 2.93 mSv, respectively.

Based on these results, a plume EPZ would be confined to the on-site area. Off-site emergency response planning for the CAP200 could be simplified accordingly.

3.3.5. Tsinghua University Institute of Nuclear and New Energy Technology, China

Research performed at the Tsinghua University Institute of Nuclear and New Energy Technology (INET) was focused on the High-Temperature Gas-cooled Reactor Pebble-bed Module (HTR-PM). The HTR-PM reactor, constructed on the site of the Shidao Bay Nuclear Power Plant in China's Shangdong Province, features two 250 MW(th) HTGR modules that drive a single 210 MW(e) steam turbine. The primary areas of research are as follows.

3.3.5.1. Establishing a methodological framework for determining small modular HTGR EPZs

Methods for determining the accident sequences, calculating the emergency basis release category and evaluating the accident source term for the HTR-PM were explored.

Forty-three accident release categories were identified as relevant to the HTR-PM EPZ based on the plant Level 2 probabilistic safety assessment (PSA). From this result, five enveloping release categories were selected as the basis for nuclear emergency analysis. These were labelled P2 (small loss of coolant accident in the primary loop), P4 (large loss of coolant accident in the primary loop), SG61 (steam generator tube rupture), SG91 (steam generator header rupture) and T4 (pressure relief in the primary loop).

Two ultrafast methods were also proposed for calculating a three dimensional dose rate field from arbitrary radionuclide distributions. The first method was based on the fast Fourier transform (FFT) [31], which operates on a Cartesian grid and accelerates the computational speed by several orders of magnitude without loss of accuracy. The second method, based on the non-uniform FFT (NFFT) [32], operates on non-equispaced and arbitrarily shaped grids, and can accelerate the computational speed by two orders of magnitude. Both methods outperform the tabulated methods and semi-infinite cloud based methods in field experiment validation.

3.3.5.2. Establishing acceptance criteria for small modular HTGR EPZs

As part of this work, NUREG-0396 [20] and the Chinese standard GB/T 17680.1-2008 [33] were reviewed. Discussions took place with the National Nuclear Safety Administration (Chinese regulatory) about the acceptance criteria for HTR-PM considering the technical features of the reactor.

The methodology for determining the EPZ around the HTR-PM follows the guidelines for contemporary PWRs in GB/T 17680.1-2008 [33], but it has been adjusted to be consistent with the features of HTGRs. The emergency basis release category is proposed for source term calculations according to the method described in NUREG-0396 [20].

3.3.5.3. Demonstrating technical feasibility of simplified off-site response for small modular HTGRs

The EPZ for the HTR-PM was calculated based on the weather conditions at four Chinese nuclear sites, including the Shidao Bay Nuclear Power Plant, using the proposed methodology. The results demonstrate that the EPZ boundary is within 250 m for all four sites and it is technically feasible to simplify the off-site emergency response for the HTR-PM as compared to contemporary large NPPs.

3.3.6. Technical Research Centre, Finland

The Technical Research Centre of Finland (VTT) conducted its research in cooperation with the Korea Atomic Energy Research Institute (KAERI) and King Abdullah City for Atomic and Renewable Energy (K.A.CARE, Saudi Arabia). KAERI provided VTT with SMART core radioactive inventory and accident release information. The description here refers to all the work done by the three partner institutes, as initially planned for participation in the CRP.

The research project proceeded through the following successive phases:

- Review of prerequisites for the research activities:
  - Design of the Korean SMART reactor;
  - Site area of the SMART design;
  - IAEA Safety Standards and technical guidance for EPR;
  - Potential calculation codes for determining the release to atmosphere;
  - Potential calculation codes for off-site dose assessment.

The core radioactive inventory of SMART (generic design) and the list of accidents postulated for SMART were prepared by KAERI. The possibility of a severe accident, even with very low probability, was considered.

- Characterization of off-site release was performed based on:
  - The use of MELCOR (an integral severe accident simulation code)
  - VTT expert judgement and use of conservative release fractions.

A ‘worst case’ (deterministic) off-site release scenario was defined considering a main steam line break (MLSB) that induces a steam generator tube rupture (SGTR), leading to a severe accident.

- Consideration of site specific conditions (weather, surrounding environment, e.g. possible resuspension from sandy ground): one site in the Republic of Korea, one site in Finland and one site in Saudi Arabia (primary consideration).
- Selecting and acquiring representative weather data for:
  - Finland/Olkiluoto weather data were directly available at VTT (wind speed and direction, rain, stability), with permission from Teollisuuden Voima Oyj (TVO), Finland to use for the purpose of the CRP;
  - Republic of Korea’s selected site;
  - Saudi Arabia’s selected site weather data.
- Atmospheric dispersion and dose assessments using the following codes: ARANO, an in-house code in VTT and MACCS, the code from the United States NRC.

Post-treatment of relevant dose results comprise [34]:

- Relevant pathways of dose exposure that include inhalation, cloud shine, ground shine, bone marrow, and lungs;

— Relevant exposure duration, based on the generic criteria in GSR Part 7 [4]);
— Chosen fractiles (95% or 99.5%), at dose level, which is exceeded with low probability (5% or 0.5%) when the cases are those calculated with all available weather situations (MACCS: 'weather trials');
— Distances from the reference NPP of the possible extent of the EPZ;
— Comparison of projected dose levels with IAEA criteria for protective actions, given in GSR Part 7 [4], Appendix II, as generic dose criteria.

The suggestion obtained for an EPZ size can be compared with the simple scaling calculation by VTT (scaling down from the Olkiluoto-3 large reactor EPZ). The resulting EPZ sizes are smaller than those for a generic large power reactor (size of e.g. Olkiluoto-3).

3.3.7. Korea Atomic Energy Research Institute, Republic of Korea

The SMART design is a light water SMR with a nominal thermal power of 365 MW(th), developed at KAERI. It is proposed that the methodology to evaluate the EPZ for SMART follows the IAEA guidance and the United States NRC literature for conventional large LWRs. In addition, accident analysis tools such as MELCOR and MACCS are used. Adopting a deterministic approach in the initial stage of the study, as the 'worst case scenario', an SGTR accident induced by a MSLB, leading to a severe accident situation due to additional postulated failures of safety systems, is analysed using MELCOR. The atmospheric dispersion of radioactive materials as well as the associated projected doses are calculated based on regional meteorology and topography data, using the MACCS code, for a specific Korean NPP site. The dose projection as a function of distance for the SGTR scenario is evaluated and compared with the dose projection for the siting source term (SST) given in Technical Guidance for Siting Criteria Development (NUREG/CR-2239) [35]. Three source terms, SST1, SST2 and SST3, from [35] were applied to SMART at a location in the Republic of Korea. Since the SST1 in [35] is for a large early release containment failure sequence, the distance that ensures a maximum projected dose <50 mSv effective dose (criterion for evacuation in the Republic of Korea's Protection Action Guide [36] is ~20 km based on the mean value. For the SST2 and SST 3 source terms in [36], respectively for late containment failure and no containment failure, the distances that ensure a maximum projected dose <50 mSv effective dose are 2.0 km and 0.2 km, respectively. For the above SGTR sequence, the distance that ensures a maximum projected dose <50 mSv effective dose is between those obtained using SST1 and SST2. The selection of the accident sequence as a limiting case for the EPZ size evaluation is the first step. Since the dose to distance to determine the EPZ size is quite dependent on the accident sequence, it is seen as a good practice to consider the use of a spectrum of accidents for more realistic assessment of EPZ sizes (i.e. a probabilistic approach based on Level 2 PSA results). A risk informed approach to consider the accident scenarios as an aggregation method and also to assess the dose to distance with sensitivity analysis for uncertain parameters in weather conditions and dose assessment was followed. The geographical extent of off-site radiological consequences in case of a nuclear emergency at SMART seems to be 'in the neighbourhood' of the site boundary. Therefore, the EPZ size for SMART might be reduced. However, since there are still many uncertainties in the analysis results of Level 1 PSA and source term, those results should be refined more thoroughly.

3.3.8. King Abdullah City for Atomic and Renewable Energy, Saudi Arabia

As a cooperative work, K.A.CARE, KAERI and VTT evaluated the EPZ size of SMART for a specific Saudi Arabian candidate nuclear site.

The initial radioactive core inventory of SMART was used, and the accident sequences were selected as a ‘worst case accident scenario’ to evaluate the EPZ size based on the severe accident analysis and Level 1 and 2 PSA results for SMART. The methodology to evaluate the EPZ for a Saudi Arabian site is the same as for KAERI’s work, except for the weather conditions used for atmospheric dispersion. In addition to the MACCS code that was also used by KAERI for the Korean site, the HOTSPOT code was used by K.A.CARE for atmospheric dispersion and dose projection for the Saudi Arabian site. The weather data used show a typical desert climate condition, reflecting Saudi Arabia’s topographical features. In particular, the meteorological data for the site are collected from a weather station that was installed in 2019, including wind speed, wind direction and temperature at different elevations. The data were obtained for each second in the time period from 1 October 2019 to 1 October 2021.

The data were averaged for each hour in that time period to prepare the meteorological input file for the consequence analysis using MACCS and HOTSPOT codes.

Since the dose to distance to determine EPZ is quite dependent on the accident sequence in the deterministic approach, the probabilistic approach based on the SMART Level 2 PSA results was applied to the Saudi Arabian site. A risk informed approach to consider the spectrum of accident scenarios with the aggregation method was applied to the Saudi Arabian site to complete the estimation of the EPZ size.

3.3.9. National Research and Innovation Agency, Indonesia

The National Nuclear Energy Agency of Indonesia (BATAN), which merged into a new organization named the National Research and Innovation Agency (BRIN) in 2019, conducted research on the assessment of EPZ size for a 100 MW(e) PWR at a specific site in Indonesia. Indonesia prepared the candidate sites for constructing 10 MW(th) power reactors in the city of Serpong and in some selected sites of the Muria Peninsula (MURIA), Serang Coastal Area (SERANG) and South and West Bangka Belitung (South BABEL and West BABEL) for 300 MW(th) SMRs. To meet the regulatory requirements from the BAPETEN for licensing SMRs, the data required for sizing EPZs for SMRs were obtained. This study focused on the determination of EPZ sizing calculations and includes postulated source term and analysis of dispersion and dose projection for the selected sites. Calculation of the source term included neutronic analysis, thermohydraulic analysis, accident progression analysis and reactor safety analysis. The analysis of atmospheric dispersion and dose projection require input data, including on the source term, meteorology, population distribution, agricultural production and local farming. The EPZ size is influenced by the size of postulated accident source term and reactor safety features, as well as site topography and meteorological conditions.

The purpose of the work was to determine the size of the EPZ for SMRs in the 100–300 MW(th) power range for Muria Peninsula, Serang Coastal and Bangka Belitung Island sites. The scope of the study includes the calculation of the core inventory of 100–300 MW(th)

SMRs of the PWR type, source term evaluation of postulated accidents, atmospheric dispersion analysis and EPZ size determination. Calculations were performed using the software available in BATAN, as well as meteorological data, population distribution and other environmental data adapted to site data. This study was conducted over three years. The first year focused on calculating the EPZ size for the Serpong site considering a 10 MW(th) HTGR. The second year focused on calculating the source term for the LWR 100–300 MW(th) type of SMR. The third year focused on the determination of the EPZ size for the three candidate sites for 100–300 MW(th) LWR type SMRs.

The assessment approach in this work was based on atmospheric dispersion calculation, using PC COSYMA. Data inputs, such as meteorological data, population distribution and other environmental data adapted to site data, were obtained from National Statistics Agency and National Meteorology, Climatology and Geophysics Agency. Calculation outputs consist of source terms, annually dominant wind directions and dose for age categories within a certain radius from the reactor.

3.3.10. Soreq Nuclear Research Centre, Israel

The SMR design considered by Soreq Nuclear Research Centre for its research in the CRP was the NuScale design, that has built-in features that are claimed to prevent the possibility of accidents of various types and, if one happens, to minimize or even eliminate environmental damage. For its work in the CRP, Soreq Nuclear Research Centre considered that one of the main significant threats posed by a hostile activity against a NuScale NPP would be a direct hit by a ground penetrating weapon (GPW). Following the development of the basic methodology for computing source terms and demonstrating/testing it on a few synthetic simplistic scenarios, Soreq Nuclear Research Centre considered some more elaborate situations, taking into account some specific features of the NuScale NPP design. Perhaps the most severe scenario is when a GPW penetrates the building’s roof, falls and explodes in the pool. The overall effect of an underwater explosion depends on various parameters, including depth, the size and nature of the explosive charge, and the presence, composition and distance of reflecting surfaces, the water surface and thermoclines. Ship building/naval warfare literature, as well as publications on safety analysis of underwater oil pipelines, can provide relevant technical references in that area. An underwater explosion is characterized by a blast wave followed by an initially expanding and later pulsating ‘bubble’ of gaseous explosion products. The outcome of each particular scenario depends mainly on the module’s proximity to the explosion, on the depth of the explosion and, of course, on the charge size.

The overall purpose of the research was to investigate the integrity of the reactor pressure vessel and containment building in case of a GPW hit.

3.3.11. Japan Atomic Energy Agency, Japan

The Japan Atomic Energy Agency (JAEA) outlined current regulations on nuclear emergency planning in Japan. JAEA pointed out the current challenging situation, seen as a lack of methodologies for determining the size of a PAZ, taking into account the specific features of SMRs and criteria for emergency action levels for HTGRs in domestic and international standards. To solve the situation, two research objectives were set by JAEA:

— To develop a methodology to determine the size of a PAZ and apply it to modular

HTGRs to confirm that the size of a PAZ would not exceed the size of the site;
— To develop a process for determining criteria with support from PSA and apply it to HTGRs to make use of the criteria.

The developed PAZ evaluation methodology was applied to Gas Turbine High Temperature Reactor 300 (GTHTR300) as a case study.

3.3.12. Toshiba Corporation, Japan

MoveluX is a 10 MW(th) class multipurpose microreactor whose concept uses heat pipes for primary core cooling and sodium as the working coolant. The heat pipe provides passive safety features as well as system simplification. MoveluX uses low enriched uranium fuel of <4.99 wt%. A severe accident scenario was identified to evaluate the source term used in Toshiba's research for the CRP. The importance of a modernized EPZ for SMRs and advanced reactors was discussed. Toshiba worked on suggesting criteria for appropriately sizing an EPZ for an SMR, based on mitigation capability for a core melt due to inherent safety aspects. As an example, a suggested EPZ size for the Toshiba 4S (Super Safe, Small and Simple) SMR was introduced.

3.3.13. Joint Research Centre, European Commission

The European Commission Joint Research Centre (JRC) performed the following tasks for the project:

— Review of the existing EPZ definitions and calculation methodologies in various countries, as well as guidelines and recommendations at international level;
— Analysis of national legislation on EPZ and regulatory acceptance criteria for the determination of EPZ sizes;
— Analysis of SMR technologies from an EPR and EPZ standpoint with the aim of identifying those particular aspects differing from conventional large NPPs;
— Assessment of EPZ methodologies applied to SMRs;
— Development of a technology neutral methodology for the accident selection and source term calculation suitable for different national acceptance criteria (deterministic, risk informed, risk based) and different state of the art and available software calculation and simulation tools;
— Identification of underlying driving criteria for the definition of EPZ areas with particular attention given to the innovative aspects brought by SMRs;
— Analysis of the existing approaches and related regulatory acceptance criteria for the determination of EPZ sizes;
— Assessment of the impact of establishing several EPZ sizes for several SMR modules against having one EPZ for a large NPP featuring a similar overall output power.

3.3.14. Pakistan Atomic Energy Commission, Pakistan

The research activities conducted by Pakistan Atomic Energy Commission (PAEC) were as follows:

— For integral PWRs of 50 MWe and 100 MWe, EPZs were estimated using IAEA dosimetric criteria and dose to distance methodology considering source term (based

on scaled down source term for a 1100 MW(e) Gen III+ PWR), environmental releases of radioactive materials, meteorology and radiological doses for different exposure pathways (i.e. cloud shine, inhalation, ground shine). Detailed analysis was performed considering site specific data for the two selected sites using the -MACCS code.
— Sensitivity cases were performed for inland and coastal sites, with different power levels and release fractions.
— As the decision on the size of EPZs represents a judgment on the extent of detailed planning that has to be performed to ensure an effective response. Therefore, analysis was performed for the optimization of protective actions (i.e. evacuation, sheltering, iodine thyroid blocking (ITB)) considering the generic criteria given in [11]. Effective dose was calculated for a period of seven days and is the sum of doses from external exposure to radioactive materials in the plume, external exposure to radioactive materials deposited on the ground and internal exposure from inhalation of radioactive materials in the plume. The equivalent dose to the thyroid was calculated from inhalation of radioactive materials. Dose reduction factors were considered for various pathways and shielding structures. The MACCS computer code was used for iPWR-50 analysis using the site-specific parameters for coastal site. The release time frame modelled in this assessment was the first seven days after an initial release as per the technical guidance in [11] and modelled as the early phase in the MACCS code.

3.3.15. Tunisian Company of Electricity and Gas, Tunisia

The Tunisian Company of Electricity and Gas (STEG) conducted its research considering an ACP100 (100 MW(e)) type of SMR that would be located in a pre-identified site in Tunisia, following a deterministic approach. After a bibliographical review, STEG worked on the basis of one accident scenario, namely a small break loss of coolant accident (SBLOCA), for which the source term was scaled down from a large NPP based on a power ratio. The source term was divided into three groups of radionuclides for ease of use with the atmospheric dispersion simulations with ADMS software, which uses a Gaussian model. Radiological consequences from the atmospheric dispersion and deposition outputs obtained with ADMS were calculated with PC-CREAM and PACE. Based on dose outputs and non-technical considerations, STEG suggested an EPZ size.

3.3.16. Corporate Risk Associates Limited, United Kingdom

Corporate Risk Associates Limited (CRA) collaborated with Imperial College London on several projects related to the assessment comparison of societal consequences between large conventional NPPs and SMRs. All these projects applied Level 3 PSA as the tool for the quantification of the consequences linked to different plant sizes and technologies. The selected figures of merit for the assessment of societal consequences were the following:

— Deterministic dose effects;
— Stochastic dose effects;
— Impacts on land and population.

First, the research conducted by CRA studied the impact in terms of societal consequences of a pure scale plant reduction, assuming that the technology of both plants, large and SMR, is the same. The goal was to compare a nuclear fleet made up of a single large plant against several SMR sites featuring the same output energy. Should power output maintain a linear relationship with source term and should source term size display a linear relationship with consequences, the accident frequency of an SMR would be inversely proportional to the number of reactors deployed. Hence, if five SMRs were to be deployed, their individual accident frequency would need to be a fifth of that for an equivalent single reactor. Thus, this project aimed to explore whether or not the relation between consequences and source term size does in fact display this linearity.

The consequence parameters examined within the project are the deterministic and stochastic health effects as well as the area size and population within the DEPZ in the United Kingdom. The project achieved this via the use of PACE, a new software package that incorporates the advances in the field of nuclear safety and atmospheric dispersion in the industry from its predecessor COSYMA.

Second, the project compared a large conventional plant against an advanced SMR design. The resurgence of interest in the MSR as a novel, advanced reactor concept, characterized by the use of molten salt as the coolant, led to an interest in the benefits it might hold over more conventional reactors. Interest in its safety advantages led to this project, where PACE, which was provided by Public Health England, was used to evaluate the harm caused in the event of an accident scenario.

The parameters evaluated for the project were the deterministic and stochastic health effects induced by the selected source term, as well as the affected area and population within a dose limit. The project used a source term provided by an MSR vendor and compared it with the RC-602 basemat failure release category for the United Kingdom's future European Pressurized Reactor.

Third, the project aimed to compare the prospect of a nuclear fleet consisting of either only large reactors or only small reactors in the safety case for the United Kingdom's six newly proposed nuclear new build sites. To do so, PACE was used to model the atmospheric dispersion of radioactive material in an accident scenario, and to calculate the associated radiological consequences. The report also considered the optimization of the PACE runs by reducing the source term input data and sensitivity analysis of the results to meteorological condition time frames.

3.3.17. Texas A&M University, United States of America

Research at Texas A&M University (TAMU) explored important aspects of determining the EPZ for modular HTGRs, particularly:

- The gas mixing inside HTGR buildings;
- The concentration and timescale of the gas mixtures that could be released into the atmosphere through the reactor building when accidents occur;
- The phenomena governing the transport of fission products.

This work made use of the Reactor Building Facility at TAMU. This facility was designed to:

- Identify key system parameters important during depressurization accidents;

— Study the reactor building response under depressurization and air ingress scenarios;
— Produce experimental data to support the validation of advanced computation tools.

The experiments that were conducted provided important information about the gas mixing inside the modular HTGR building, as well as the the concentration and timescale of the gas mixtures that could be released into the atmosphere, during hypothetical accident scenarios such as small and moderate size primary breaks of pressure boundaries. Specifically, the main flow paths of helium and air were identified inside the reactor building, and the event times for the main phenomena expected to occur during the air refill phase were estimated. This new information elucidated the important phenomena inside the reactor building and their timing, and is useful for the validation of system level codes and CFD codes that support the estimation of appropriately sized EPZs for SMRs.

3.3.18. Argonne National Laboratory, United States of America

Source term is a vital metric use to establish containment/confinement design, site boundary, and EPZ. It is a major focus of advanced reactor licensing, and the United States NRC has stated an expectation for advanced reactor vendors to present mechanistic assessment of potential source terms in their licence applications.

In 2014, the United States Department of Energy (DOE) funded an investigation to link advanced reactor research activities to key licensing requirements and challenges, beginning with a review of possible licensing impediments for SFRs. The noted issues included code and model qualification, regulatory treatment of sodium fires, seismic isolator qualification and database quality. The development of mechanistic source term (MST) was the only issue with high regulatory importance and possible long lead time to closure. From 2015 to 2019 the DOE funded research in the area of SFR MST at national laboratories and universities.

Argonne National Laboratory (ANL) created the MST development pathway considering only the metal alloy fuelled, pool type SFR, but the methodologies should be applicable to other types of advanced reactors. The contributions from ANL are summarized in the three publicly available reports described below.

Regulatory Technology Development Plan: Sodium Fast Reactor — Mechanistic Source Term Development (ANL-ART-3) [37] was the initial survey report that identified gaps in knowledge concerning radionuclide release from failed metal fuel into sodium, and a lack of a cohesive radionuclide transport modelling computational tool. The conclusion was that a trial MST calculation was necessary.

Regulatory Technology Development Plan: Sodium Fast Reactor — Mechanistic Source Term — Metal Fuel Radionuclide Release (ANL-ART-38) [38] presents a review of the current state of knowledge of radionuclide release from failed metal fuel and describes the development of realistic radionuclide release fractions to the primary sodium. These release fractions were identified for all of the radionuclides listed in Accident Source Terms for Light-Water Nuclear Power Plants (NUREG-1465) [39] (alternative LWR source term).

Regulatory Technology Development Plan — Sodium Fast Reactor: Mechanistic Source Term Development — Trial Calculation (ANL-ART-49) [39] summarizes a trial MST analysis. The goal was to explore all pertinent phenomena associated with radionuclide release by examining two types of transient scenarios: a loss of primary system flow with scram and degraded heat removal, and a severe transient overpower without scram and reduced inherent reactivity feedback. Both scenarios were beyond the design basis of the reference SFR design.

The key conclusions were that a metal alloy fuelled, pool type SFR MST calculation was possible with currently available tools and models, but gaps do exist in the models and data for some phenomena. These gaps result in higher uncertainties or require the use of conservative assumptions, and could potentially impact on design decisions relying on source term analysis, including site boundary and EPZ.

3.3.19. Pittsburgh Technical, United States of America

For Pittsburgh Technical, the current NRC rulemaking is believed to support correctly sized EPZs, but a technical basis is required to support a risk informed approach. Pittsburgh Technical has been working with other Electrical Power Research Institute (EPRI) partners to ensure that developments are:

- Aligned — consistent methodologies are developed and there is agreement between stakeholders;
- Applicable — work is incremental to providing the technical basis that underpins the industry’s position;
- Ahead — relevant projects are proactively identified to support SMR development and deployment.

The research contributions of Pittsburgh Technical toward the development of this technical basis can be separated into the categories of outside containment and inside containment.

3.3.19.1. Outside containment

One objective was to improve atmospheric dispersion methods, models and tools. To that end, a technical basis was developed to support use of the AERMOD computer code for NPP applications. The AERMOD code is permitted and used by the Environmental Protection Agency (EPA) in the United States of America. It includes additional dispersion mechanisms not previously featured in nuclear accident atmospheric dispersion modelling tools, such as improved handling of building wake effects and stack tip downwash.

Another objective was to improve evacuation time estimates. A new consequence estimation tool was developed in Microsoft Excel that maps population movement against doses estimated from the RASCAL code to determine dose consequences for the population. The new tool allows variation of critical parameters such as time between event occurrence and notification and the direction of cohort movement. The dose consequence was determined based on parameters such as message timing, message content and quality, cohort adherence or compliance, daytime and nighttime population variations, and different evacuation and sheltering methods.

3.3.19.2. Inside containment

This research investigated decontamination factors. The purpose was to develop validation data that would support the use of containment decontamination factors based on the natural phenomena expected in the relatively large surface area to volume ratio of SMR containments. The previously available correlations were based on large LWRs, and the SMR specific correlations were expected to provide reduced design basis and beyond design basis accident source terms.

Pittsburgh Technical constructed the SMR Test Containment Vessel to represent multiple water cooled SMR containments and thermohydraulic conditions. The apparatus was designed with variable volume and surface area at a typical height and diameter, and could create various immersed/non-immersed, cold wall/hot wall, and gas cavity temperatures based on the conditions of interest. Measurements were made of impaction velocimetry, particle size, steam condensation and pressure.

Improved non-invasive measurement techniques needed to be developed for the experiment. Prior attempts used coupons that required post-test sampling, potentially introducing errors. Particle concentration measurements with a fluorometer were not practical because of the high temperature environment. An alternative non-invasive technique, based on laser particle imaging, was therefore employed.

Phase 1 of the effort identified diffusiophoresis, thermophoresis and gravitational settling as important aerosol removal phenomena. However, the Phase 2 effort identified and quantified the significance of convective flow as the dominant deposition mechanism. This finding was significant because, although convective flow simulation capabilities exist with tools such as CFD, there is no evidence of its credit as an aerosol removal mechanism in current accident analysis codes such as MELCOR and MAAP. This mechanism might account for up to 50% of the particle deposition observed in the experiment. Based on the experimental results, a representative decontamination factor for an integral pressurized water reactor was estimated at 19.7, which is significantly higher than the value of 1.2 commonly attributed to large LWRs.

4. STEPS FOR THE DETERMINATION OF EMERGENCY PLANNING ZONE SIZE AND EMERGENCY PLANNING DISTANCE FOR SMALL MODULAR REACTORS

4.1. HAZARD ASSESSMENT

Paragraph 4.20 of GSR Part 7 [4] states that:

> "The government shall ensure that for facilities and activities, a hazard assessment on the basis of a graded approach is performed. The hazard assessment shall include consideration of:
>
> (a) Events that could affect the facility or activity, including events of very low probability and events not considered in the design;
> (b) Events involving a combination of a nuclear or radiological emergency with a conventional emergency such as an emergency following an earthquake, a volcanic eruption, a tropical cyclone, severe weather, a tsunami, an aircraft crash or civil disturbances that could affect wide areas and/or could impair capabilities to provide support in the emergency response;
> (c) Events that could affect several facilities and activities concurrently, as well as consideration of the interactions between the facilities and activities affected;
> (d) Events at facilities in other States or events involving activities in other States."

Sections 4.1.1–4.1.6 provide some background of hazards to be considered for the conduct of hazard assessment. This is not intended to be an exhaustive list.

4.1.1. Design basis accidents and beyond design basis accidents

Institutes identified and considered hazards[1] that served as inputs for design basis accidents (DBAs) and beyond design basis accidents (BDBAs), including design extension conditions (DECs)[2].

A DBA is "a postulated accident leading to accident conditions for which a facility is designed in accordance with established design criteria and conservative methodology, and for which releases of radioactive material are kept within acceptable limits" [12].

DBAs are accidents that serve as basis for demonstrating the safety of the facility, designing safety systems and equipment as well as accident management procedures. The operator demonstrates to the regulator that the frequency of occurrence of those accidents is made low enough to be acceptable (as per national regulations). The CRP included scenarios involving DBAs that lead to radiological consequences off-site.

A beyond design basis accident is a "Postulated *accident* with *accident conditions* more severe than those of a *design basis accident*" [12]. A beyond design basis accident that

[1] Hazard is defined as "the potential for harm or other detriment, especially for *radiation risks*; a factor or condition that might operate against *safety*" [12].

[2] Design extension conditions are defined as "Postulated *accident conditions* that are not considered for *design basis accidents*, but that are considered in the *design process* of the *facility* in accordance with best estimate methodology, and for which *releases* of *radioactive material* are kept within *acceptable limits*" [12].

involves significant core degradation is a severe accident, and these have been a topic of past emphasis. For example, for PWR types of SMRs, the CRP included scenarios involving:

- The loss of the main feedwater system with the failure of the passive heat removal system and condensate makeup tank (CMT); or
- A medium primary break LOCA with CMT failure.

Potentially combined with:

- An early containment failure; or
- A bypass of containment or a containment isolation failure.

4.1.2. Combination of a nuclear emergency with a conventional emergency

In GSR Part 7 [4], a nuclear or radiological emergency is defined as:

> "An emergency in which there is, or is perceived to be, a hazard due to:
>
> (a) The energy resulting from a nuclear chain reaction or from the decay of the products of a chain reaction;
> (b) Radiation exposure".

Conventional emergencies include fires, chemical emergencies, explosions (non-security related), aircraft crashes and any other emergency not related to radiation (e.g. pandemic), which would be expected to be included in national all hazards emergency plans.

Past experience has shown that nuclear emergencies at NPPs can have severe radiological or non-radiological health consequences for the public. When combined with non-conventional emergencies, the consequences for health might be even greater due to there being more challenging conditions for the implemention of public protective actions in a timely manner.

The topic of combined emergencies was not considered in the CRP.

4.1.3. Emergency triggered by a nuclear security event

One institute that was part of the CRP considered emergency scenarios triggered by events of a nuclear security nature, namely, the explosion of a GPW inside the containment building. The explosive charge of the GPW was the same in all scenarios, but the trajectory of the GPW and the location of the explosion and its subsequent damage varied among the set of scenarios. Another institute that was part of the CRP considered attacks on locally stored spent fuel. While reactors are potentially less vulnerable to attacks, especially if they are below grade, spent fuel is commonly stored in pools nearby the reactor, but not necessarily in containment, or in dry casks that in some cases are outdoors. Nuclear security-driven events could have consequences that are significantly different from reactor accidents if the explosive charge is able to directly aerosolize spent fuel [40]. Security-driven events might be more common if SMRs are deployed within conflict zones [41].

4.1.4. Emergency affecting SMR(s) collocated with existing operating NPPs

Several NPP countries that plan to deploy SMRs envisage to do so first on sites where NPP reactor units are already in operation. Even though EPR arrangements for SMRs deployed on such sites might be encompassed by EPR arrangements already in place for the existing reactor units, this area would benefit further investigation.

4.1.5. Emergency affecting more than one SMR module

Many proposed SMR deployments include the construction and operation of more than one SMR module at a given site. A multi-module deployment at a given site is not similar to an existing 'large' NPP site with several reactor units built. As highlighted by the IAEA's SMR Regulators' Forum [2]:

> "The concept of multi-modules is specific to SMRs, and thus should be considered as an important safety issue to be investigated, particularly in comparison with current practices on nuclear safety for large reactors."

And further:

> "The multi modules/units aspect of SMRs should be considered in the internal hazard safety assessment, particularly in terms of:
>
> - propagation of internal hazards from one module to another (e.g., fire propagation)
> - the impact of operating activities of one module on the risk of internal hazard of other modules (e.g., the risk of load drop due to the refuelling of one module)" [2].

Some institutes that participated in the CRP opened perspectives for future work in the area of EPR for multi-modules; however this topic was not primarily addressed as part of the project.

4.1.6. Conclusions on the main hazards considered throughout the CRP

Overall, most hazards considered by institutes in the CRP were internal hazards, that is, hazards related to the loss or failure, or series of losses and failures, of major safety systems or components.

Based on the hazards considered, national/local considerations and the resources available, each institute built its own set of accident scenarios or sequences, in terms of the number of scenarios or sequences, their duration, their triggering events and the severity of their projected radiological consequences.

4.2. IDENTIFICATION OF ACCIDENT SCENARIOS

The approach toward identifying accident scenarios used to determine the EPZ can be classified as either deterministic or probabilistic. The contributions made to the CRP are summarized in the following subsections. Table 2 summarizes which approach was used by which contributor. This list includes only the research projects that produced an explicit EPZ determination. Some contributors performed individual pieces of analysis that could be classified as deterministic or probabilistic in nature, but these were excluded from the table if the analysis was not part of an overall method to determine an EPZ. Some contributors are listed in both categories because they performed multiple types of analyses.

TABLE 2. APPROACHES USED BY DIFFERENT CRP CONTRIBUTORS

	Approach to accident sequence identification	
	Deterministic	**Probabilistic**
Contributor	CNNC/CNPE (China)	CNNC/CNPE (China)
	BATAN (Indonesia)	SNERDI (China)
	JAEA (Japan)	INET Tsinghua University (China)
	Toshiba (Japan)	KAERI (Republic of Korea)
	KAERI (Republic of Korea)	K.A.CARE (Saudi Arabia)
	PAEC (Pakistan)	
	K.A.CARE (Saudi Arabia)	
	STEG (Tunisia)	

4.2.1. Deterministic approach

A deterministic analysis is an "Analysis using, for key parameters, single numerical values (taken to have a probability of 1), leading to a single value for the result" [12]. This contrasts with a probabilistic analysis, where key parameters might have many different values, each with an associated probability, which leads to a distribution of possible outcomes. With regard to event identification, the deterministic approach refers to the selection of an event or sequence where the consequence will subsequently be assessed to determine the EPZ. The participating institutes chose an event using a best estimate or conservative rationale that is based on expert judgement and knowledge of the underlying phenomena. Here, 'best estimate' refers to the most likely prediction of an uncertain behaviour, whereas a 'conservative' estimate is intended to be limiting or bounding based on current knowledge. This definition does not preclude the consideration of multiple events that have been assessed deterministically.

The contributions to this CRP included consideration of events ranging from design basis to beyond design basis. The rationale for selecting these events varied significantly among the contributors. Some were assessed according to national regulatory requirements for deterministic safety analysis of nuclear power plants. Some postulated events were investigated simply because study of the underlying phenomena was the focus of the research contribution. However, in the majority of examples provided by the contributors, events were argued to be bounding with respect to source terms or off-site consequences and thus represented a suitable basis for determining the EPZ.

Several participants noted experience with Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants (NUREG-0396) [20]. That publication describes the concept of a Class 9 accident that includes either total core melt leading to degradation of the containment boundary, or gross fuel cladding failure or partial core melting with an independent failure of the containment boundary. Such an accident in a large LWR has been the basis of a 10-mile EPZ for plume exposure in the United States of America. In some cases, the contributor claimed that it would be unreasonable to assume core melting or gross fuel cladding failure because the passive or inherent safety features of a particular SMR technology precluded the possibility (e.g. in the case of HTGRs). Other participants assumed the occurrence of events including gross fuel cladding failure purely for the purpose of determining the EPZ without suggesting an initiating event by which such a failure could occur.

Multiple contributors described the deterministic assessment of two categories of DBA: an infrequent accident and a limiting accident. Infrequent accidents were not expected to occur over the lifetime of a single reactor, but might be expected when building multiple reactors, with frequencies ranging from 10^{-2} to 10^{-4} per reactor year. Limiting accidents were not expected over the lifetime of all reactors, with frequencies ranging from 10^{-4} to 10^{-6} per reactor year. Different dose limits were defined at the site boundary for infrequent and limiting accidents. According to these contributors, these DBA were identified deterministically, but the spectrum of BDBAs were identified probabilistically (Section 4.2.2). Multiple contributors that analysed water cooled SMRs identified a fuel handling accident (FHA) as the limiting or bounding DBA for that technology. Other contributors identified loss of containment isolation accidents as a BDBA to explore how the function of passive safety features could be impaired. Other contributors analysed DBAs such as SGTR or SBLOCA, without claiming that such an accident was necessarily bounding. In these latter cases, it might be that the size of the EPZ was assumed to be bounded by a different BDBA, so justification that a certain event was bounding for all DBAs was not necessary. Bounding BDBAs that were assessed for water cooled SMRs by the contributors were mostly internal events (e.g. loss of main feedwater coincident with passive residual heat removal system and CMT failure, large break loss of coolant accident (LBLOCA) with fuel melting, loss of water in the operating pool). In one case, a contributor postulated the bounding event to be the impact of a GPW in the reactor building and the operating pool of a multi-module plant.

For an HTGR, one contributor identified SGTR as the limiting DBA. Multiple contributors suggested that the bounding BDBA in HTGRs involved a break in the primary helium circuit, air ingress into the core region and oxidation of the coated particle fuel.

Participants that investigated liquid metal cooled fast reactors, including SFRs and LFRs, postulated bounding BDBAs involving severe reactivity excursions with failure to scram. One participant postulated a Class 9 failure (according to NUREG-0396 [20]) with gross fuel cladding failure specifically for the purposes of determining the EPZ.

One participant postulated a bounding event in a molten salt reactor that included a failure of the primary vessel/circuit coincidence with the ingress of water into the reactor cavity (e.g. caused by a beyond design basis earthquake and flooding), resulting in the interaction between molten fuel salt and water.

In summary, the majority of events that were assessed deterministically by the contributors were accidents involving the failures of systems, structures, or components that were argued by the contributor to be bounding in some way, either bounding the realm of DBAs or bounding all accident conditions up to BDBAs.

4.2.2. Probabilistic approach

The probabilistic approach, in contrast to the deterministic approach, recognizes that key parameters might have multiple different values, each with an associated probability, that together result in a distribution of possible outcomes. The consideration of multiple states and outcomes is often manifest in a PSA (also sometimes called a probabilistic risk assessment (PRA) in certain Member States), defined as "A comprehensive, structured approach to identifying *failure scenarios*, constituting a conceptual and mathematical tool for deriving numerical estimates of *risk*" [12]. Submission and acceptance of some level of PSA is a common requirement for licensing a NPP in many jurisdictions.

The IAEA Nuclear Safety and Security Glossary [12] states:

"Three levels of PSA are generally recognized:

— Level 1 comprises the *assessment* of *failures* leading to determination of the frequency of fuel damage.
— Level 2 includes the *assessment* of *containment* response, leading, together with Level 1 results, to the determination of frequencies of *failure* of the *containment* and *release* to the *environment* of a given percentage of the reactor core's inventory of radionuclides.
— Level 3 includes the *assessment* of *off-site* consequences, leading, together with the results of Level 2 *analysis*, to estimates of public *risks*".

From these definitions, the Level 2 PSA results inform the selection of events that are used to define the EPZ. Because a Level 3 PSA considers off-site consequences and risks, the results go beyond the mere identification of events. Methods for Level 3 PSA are described in Section 4.3.1.

Multiple contributors reported that an initial Level 2 PSA had been completed for a particular SMR and the results were used as basis for determining the EPZ. In most cases, the contributors reported that the large set of event sequences in the Level 2 PSA results were grouped into a smaller set where one event within the set bounded or enveloped all the events in the set. Each group of events was attributed a single source term, sometimes referred to as a source term category (STC), that was characteristic of the set. Examples of such groupings are all events without containment failure; all events with containment isolation failure; all events with containment bypass failure; and so on. However, this should be balanced with the lack of operational experience with SMRs, which might currently lead to very theoretical values of probabilities being attributed to some failures in the PSA.

It is noteworthy that the Level 2 PSA results contributed by some participants for water cooled SMRs took advantage of trial Chinese regulatory guidance endorsing a lower 'cutoff' event frequency. This guidance stated that a frequency of 10^{-7} to 10^{-8} per reactor year could be used as a temporary lower limit when assessing severe accidents for emergency planning purposes. The frequency of containment failure was determined to be below this cutoff, and the events assessed to determine the EPZ could thus feature an intact containment. One participant nevertheless contributed a 'what if?' analysis including different categories of containment failure (e.g. isolation failure, bypass failure, etc.) to compare directly against the result with intact containment.

As an alternative to screening out events below a certain probability threshold, one participant described a risk aggregation technique to address low likelihood, high consequence events. The described technique was based on the aggregation of conditional probabilities as described in the Nuclear Engineering Institute white paper, Proposed Methodology and

Criteria for Establishing the Technical Basis for Small Modular Reactor Emergency Planning Zone [42]. Because the aggregation was based on risk, requiring the evaluation of both frequency *and* consequences, the technique did not involve a priori screening of sequences based solely on their estimated likelihood of occurrence. Instead, a bounding distance for the EPZ was determined based on the conditional probability of the consequence (dose) exceeding a prescribed limit. Although off-site dose consequences were evaluated, the technique was described as an extension to Level 2 PSA rather than full Level 3 PSA.

Level 2 PSA results for a HTGR were also contributed. The grouping of events was significantly different from the water cooled SMR examples. This can at least partially be attributed to the fact that small HTGRs typically do not feature the same type of pressure retaining containment structure as water cooled reactors, and so the different categories of containment failure are eliminated by default. For this HTGR example, 43 categories of events in the Level 2 PSA were collected into 5 enveloping release categories:

— Small loss of coolant accident in the primary loop;
— Large loss of coolant accident in the primary loop;
— Steam generator tube rupture;
— Steam generator header rupture;
— Pressure relief in the primary loop.

Another participant noted that the WinNUPRA software was used to perform Level 2 PSA of a HTGR [42].

The Level 2 PSA results that were contributed by participants, and the events thus identified to determine the EPZ, all considered system, structure, or component failures that were either spontaneous or in response to a postulated initiating event (e.g. loss of off-site power). Dissertations on the accuracy of failure rate estimates or the magnitude of uncertainties were not included in the contributions.

4.2.3. Source terms

The calculation of the source term is strongly dependent on the identification of the accident scenarios.

The fundamental activities for the calculation of the source term are collected in Table 3 and summarized in the following sections.

TABLE 3. OVERVIEW OF SOURCE TERM CALCULATION

Field of calculation	Concept	Description
In-containment source term (ICST)	Fission product inventory	Initial quantity of radionuclides at each hazard location
	Accident conditions	Systems and barriers boundary conditions
	Release, transport, interaction and deposition	Set of phenomena affecting release, transport, decay, interaction with the surrounding materials and deposition from the initial to the final location interfacing the environment
ICST interfacing	Boundary conditions for	Thermodynamic characterization upstream of

conditions	releases of radionuclides to the outside environment	the outside environment plus closure conditions

4.2.3.1. Calculation of the inventory of fission products

This refers to the identification and characterization of the initial (at the time of the triggering event) radioactive inventory in terms of quantity and isotopic composition.

Operating NPPs feature two on-site radioactive inventory main locations: the reactor core and the spent fuel storage. Even though the inventory of the latter is larger, there are more accidental pathways for overheating fuel within a reactor to the point that fission product releases can occur. The accident progression and release kinetics for spent fuel accidents generally result in smaller representative source terms compared to reactor accident source terms, at least without significant external events to provide the driving force for releases. It is possible that the situation is different for specific SMR designs (e.g. transportable cores, sites with limited or no on-site storage) but that was not invesitgated as part of the CRP.

4.2.3.2. Accident conditions

The path followed by the fission products or activation products from the original location to the outside environment is heavily dependent on the type of accident scenario characterized by a postulated initiating event, safety systems performance and configuration, and status of the fission product barriers. This information sets the boundary conditions characterizing the accident scenario. Sections 4.2.1 and 4.2.2 focus on the different existing approaches for the selection of representative accident scenarios that would determine the source terms considered.

4.2.3.3. Release, transport, interaction and deposition

This set of calculations encompasses the chain of physicochemical events resulting in the release, transport, interaction and deposition mechanisms from the initial location to the final location interfacing the outside environment.

There exist several well established models and correlations for the mechanisms of fission product releases, mostly diffusion and convection based. Some of these models have been validated against separate effect test (SET) experiments for several radionuclides. However, some of the SMR designs, such as those for sodium cooled or molten salt reactors, incorporate novel materials and design features wherein the radionuclide transport behaviour (in terms of release, transport and deposition) has not been studied as thoroughly. As for the mechanisms of transport and deposition, several fission product barriers are put in place between the initial and final location of the radioactive material within the containment. The main transport and deposition mechanisms of fission products and activated materials include particle diffusion, coalescence, gravitation, phoresis effects, inertia currents and water scrubbing. There are several validated models that have been incorporated into both specific codes dealing with the fission products and integral system codes.

The calculation of the release, transport and deposition of the initial fission product and activated material inventory under the selected accident conditions leads to the *in-*

containment source term, which is the *inner source term* characterized as the quantity of radionuclide material, chemical composition and aerosol particle size, prior to being released to the outside environment through a set of interfacing conditions.

4.2.3.4. ICST interfacing conditions

The interfacing conditions between inside containment and the environment dictate the release rate. Interfacing conditions include:

- Thermoynamic conditions inside containment (e.g. pressure and temperature);
- Chemical composition of the atmosphere inside containment;
- Closure conditions as additional boundary conditions, for example, cross-sectional area of the flowpath to the environment and timing associated with this, filtering mechanisms and release height.

4.2.3.5. Institutes' approaches for the calculation of the source terms

The participants to this project followed different approaches for the calculation of the source terms, as reflected in Table 4.

TABLE 4. CALCULATION TOOLS OR METHODS USED BY PARTICIPATING INSTITUTES IN THIS PROJECT FOR SOURCE TERMS CALCULATION

Participating institutes	**SMR designs**	**Calculation tool/method**
China Nuclear Power Engineering Company (CNPE), Nuclear Power Institute of China, China Institute of Atomic Energy	ACP100, DHR400	Experimental and Resolution of Category 2 STDBA Source Term Method
BATAN, Indonesia	Not specified	RELAP/SCDAPSIM Mod 3.4+ART Mod 2
VTT Technical Research Centre of Finland Ltd	SMART	Direct information from PSA study/expert judgement starting from core inventory with DF/MELCOR (calculations by KAERI)
VTT Technical Research Centre of Finland Ltd	SMART	Source term based on previously used released fractions and SMART core inventory from KAERI
Nuclear Engineering, Texas A&M University, Thermal-Hydraulic Research Laboratory	SMR_HTGR	Experimental
Pittsburgh Technical LLC	Not specified	Experimental (determination of in-containment vessel decontamination factors)
Argonne National Laboratory	Small sodium fast reactors	SAS4A/SASSYS-1, CONTAIN-LMR/MELOCR
Korea Atomic Energy Research Institute	SMART 100	NUREG-1465 [39] and Reg. Guide 1.183 [43]

Directorate of Nuclear Power Engineering, Reactor — Pakistan Atomic Energy Commission	iPWR 100 MW(e)	IAEA EPR-NPP-PPA [11]
Canadian Nuclear Laboratories	30 MW(th) single unit Generic iPWR 160 MW(th)	Literature review (fraction released, initial inventory, release duration) MELCOR
Nuclear Safety Department	SMR PWR type, 100 MW(th)	NUREG-1150 [44] (release fraction) and Reg. Guide 1.183 [43]

TABLE 4. CALCULATION TOOLS OR METHODS USED BY PARTICIPATING INSTITUTES IN THIS PROJECT FOR SOURCE TERMS CALCULATION (CONT.)

Participating institutes	**SMR designs**	**Calculation tool/method**
Tunisian Company of Electricity and Gas	ACP100	Scaling factor from existing source term
Japan Atomic Energy Agency	GTHTR300	Deterministic methodology
Korea Atomic Energy Research Institute	SMART 100	NUREG-1465 [39] and Reg. Guide 1.183 [43]
CRA (with Imperial College London)	NuScale	Existing core inventory (NuScale) and release fraction
Joint Research Centre	MAAP5	Literature review

4.2.3.6. Characterization of the release

Source terms can be expressed as:

— A fraction of an initial core inventory;
— Integrated along the entire release time;
— As release rate.

The transport of radionuclides released into the atmosphere through diffusion and convection depends on various release characteristics, including:

— Release kinetics. The release kinetics, that is, the time period between the initiating event (namely the start of the accident) and the start of the release, but also the release rate and duration, has an impact on the dose projection outputs. Expressing a source term as a release rate allows to capture the kinetics of the release.
— Release height. A higher release height generally leads to more atmospheric dilution of the radioactive materials in the plume and lower projected doses. However, in some weather conditions, there can be a local maximum in the dose some distance away from the release source.
— Radionuclide characterization.

There are multiple ways of representing the release composition as implemented in integral system codes or dedicated codes:

— Fission product groups (e.g. iodine isotopes or noble gases): each features a set of radionuclides that are similar in terms of the mechanisms of release from the fuel, transport and retention.
— List of radionuclides: considering a list of specific chemical elements or radionuclides allows one to take into account important chemical properties that influence the source term.

4.3. ATMOSPHERIC TRANSPORT AND DEPOSITION

In the event of a nuclear or radiological emergency at an NPP, radioactive materials might be released to the environment through various pathways, including atmospheric dispersion, groundwater, or the nearby river or sea used for cooling. In the CRP, all participants that conducted release dispersion only considered atmospheric releases.

Information on the theoretical Gaussian, Lagrangian and Eulerian models for atmospheric dispersion can be found in Atmospheric Dispersion Models for Application in Relation to Radionuclide Releases (IAEA TECDOC-379) [45]. In particular, Fig. 2.1 in [45] provides a good illustration of the overall behaviour of effluents dispersed in the atmosphere.

4.3.1. Overview of the atmospheric dispersion models and tools used in the CRP

Figures 1 to 4 illustrate the choices of computerized calculation tools, types of dispersion modelling approaches and choices of weather data, as they were reported by the CRP participants. These charts were prepared based on the information that had been made available by the CRP participants. The amount of information that served to plot each chart varies from one chart to the next.

Most CRP participants worked on atmospheric dispersion and associated dose projection of release scenarios, using computerized dose projection tools, that is, computer codes that in some way solve the atmospheric dispersion equation and then proceed to calculate doses received from the resulting radiation source by receptor persons in the affected area. Figure 1 shows the tools used in the CRP. Note that a single participant might have used more than one tool. By far the most often referenced tool was the MACCS code, which was used by six participants.

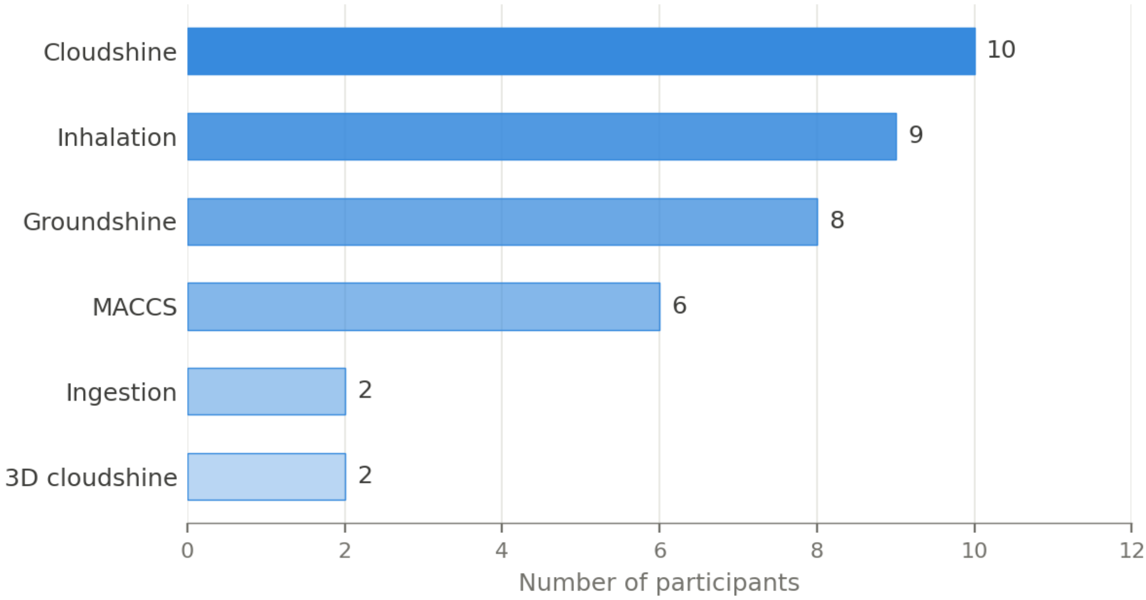

FIG. 1. Dose projection tools used by CRP participants (number of participants per tool). Note that a single participant may have used more than one tool.

The main approaches in atmospheric dispersion modelling have traditionally been classified as Gaussian, Lagrangian or Eulerian, as introduced in [45]. On the one hand, the Gaussian distribution is a simple approximation of the concentration distribution (e.g. the lateral distribution in a plume) resulting from atmospheric dispersion with certain constant/uniform assumptions. Typically, Level 3 PSA tools have resorted to Gaussian dispersion modelling because it is computable with very low central processing unit (CPU) resources and so facilitates calculating a large number of dispersion scenarios. On the other hand, Gaussian results are more approximative than results obtained with Lagrangian or Eulerian models, but this might be good enough when working on a large number of cases. Lagrangian models usually implement tracking of the trajectories of pieces of the radioactive cloud (pollutant particles/gas parcels) and the concentration will result from their superposed influence. For radioactive releases from ‘point’ sources (e.g. an NPP stack), the Lagrangian approach is higher fidelity than the Gaussian approach, at the cost of requiring greater CPU resources. Eulerian models calculate the movement of concentration between grid cells among which the calculation domain (area/volume considered) is divided. Eulerian models are therefore usually best of use for ‘area’ sources (not point sources). Lagrangian and Eulerian models are usually comparable in terms of CPU resources required and fidelity.

Figure 2 provides an overview of the atmospheric dispersion models that were used by CRP participants. Most tools used implement Gaussian models. Some of them (MACCS, PACE, VALMA) also offer a Lagrangian model option (with trajectories coming from HySplit, NAME or SILAM, respectively), but even then most CRP participants instead used the simple and fast running Gaussian modelling. Both Lagrangian and Eulerian models are available in NAME and SILAM. Furthermore, SNERDI reported having developed a CFD approach for atmospheric dispersion and dose projection. Besides, a single code tool might provide options to use more than one atmospheric dispersion model. In particular, some tools offer a choice between Gaussian and Lagrangian, but in practice most CRP participants used the Gaussian model. Further, it is worth noting that a Gaussian model might include several ‘sub-models’, for example, puff, plume, segmented plume and PGM.

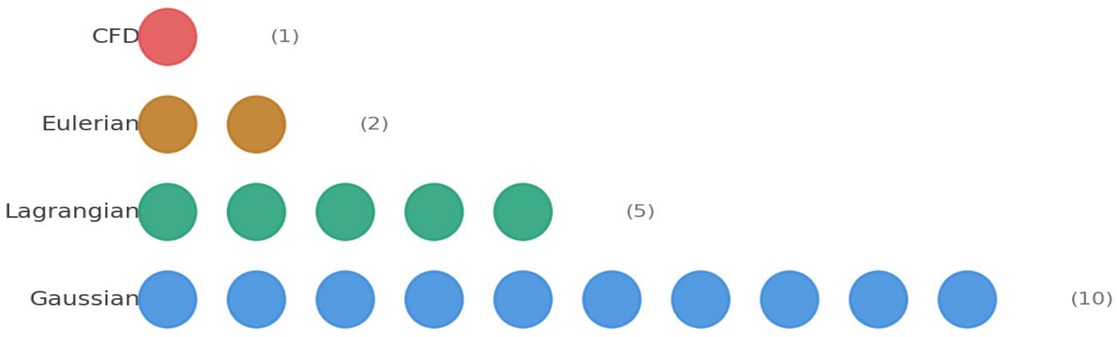

FIG. 2. The three main approaches of atmospheric dispersion modelling (Gaussian, Lagrangian, Eulerian), and their availability among the dose projection tools used.

Any computation of atmospheric dispersion is dependent on the state of the atmosphere, as reflected in the weather data. These are generally a function of spatial location and time. Weather data for dispersion might come directly from measurements, or through some kind of weather modelling, which means the generation of so called analysed fields from measurements and possibly also prediction into future time points. The CRP participants almost exclusively used measured weather data. Some exceptions were CNL reporting the use of the CALMET meteorological analysis preprocessor and VTT reporting using data from the European Centre for Medium-Range Weather Forecasts.

Figure 3 shows the distribution of the CRP participants’ use of weather cases, selected from their weather data, classified as realistic, average or conservative. Most participants have reported using realistic weather, which means a time series of site weather mast measurements spanning one year or longer, enabling a probabilistic study. Some participants (e.g. KAERI, JAEA) mentioned the use of ‘site average weather’ (i.e. a single weather condition), which can be regarded as realistic or might possibly omit important out of average/non-typical conditions, depending on the climate of the country/site location. Another distinction to be made between weather data choices is realistic versus conservative. Because the dispersed concentration results are used as the source of radiation, ‘conservative’ means those weather conditions that are likely to lead to higher individual radiation doses. Such conditions typically have stable atmosphere, low wind speeds and rain that generates wet deposition on the surfaces of ground, vegetation and buildings. Conservative weather might be found from the available weather data explicitly (by manually picking e.g. work by CNPE) or implicitly by looking at all the doses from all the weather cases, and considering a subset (e.g. 0.5% or 5%) of the cases that led to the highest doses. In practice, ‘realistic’ usually means that a large number of weather cases were used, which might be more representative than a single (average or most frequent) weather case. Using conservative weather might mean a manually tuned single weather case, or a subset of the cases that led to the highest doses.

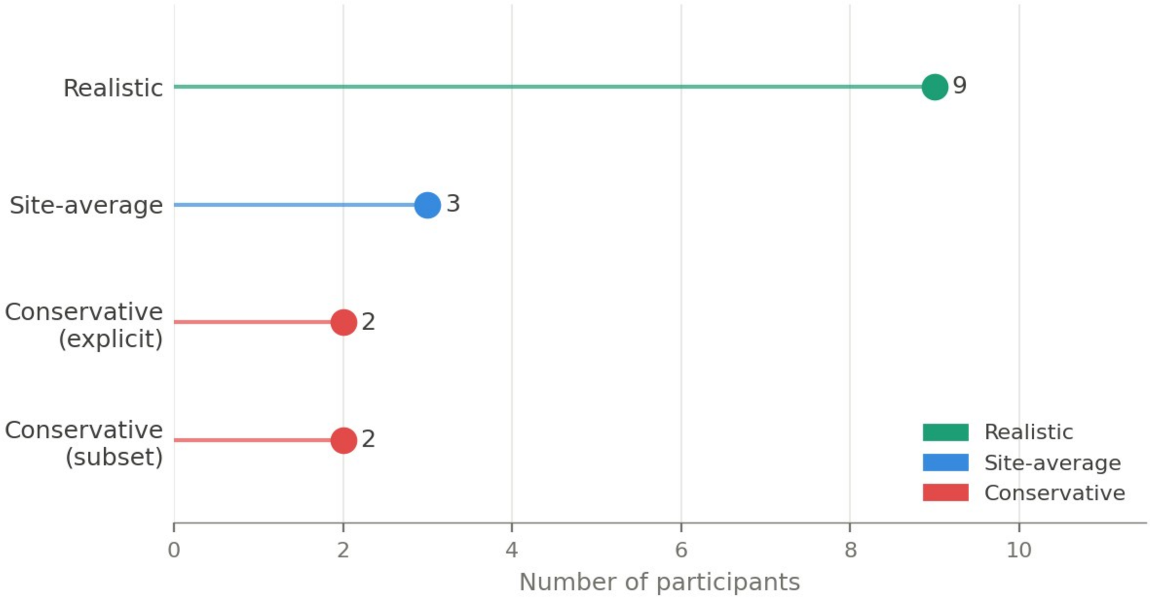

FIG. 3. Basis on which CRP participants selected weather cases for atmospheric dispersion modelling at their respective sites.

Weather data time span mainly refers to the time series used in probabilistic studies, but might also affect the choice of average/most frequent/representative weather conditions used by some participants. As mentioned above, CRP participants used almost exclusively mast measured weather data. The locations mentioned by participants included Atucha in Argentina, Chalk River Laboratories in Canada, Olkiluoto in Finland, the Republic of Korea's coast and Saudi Arabia's desert. Most NPP sites have a weather mast (in some cases more than one) with measurement devices at two or more different heights. The measurement results used by the participants' code tools were typically wind speed and direction, rain and atmospheric stability class. However, stability cannot be measured directly, but is usually computed from parameters, including vertical temperature differences and wind direction variations. Figure 4 shows the time ranges of weather measurement data used for the EPZ aimed dispersion calculations. Most CRP participants used weather measurement time series over one year or 8760 hourly measurements. This might be related to the use of the MACCS code, which was used by most participants that performed atmospheric dispersion, and accepts one year at a time in the weather data input file. The longest weather measurement time series duration reported was three years; however, most participants did not share the information on time span. Furthermore, the potential impact of climate change on historical data might need to be considered.

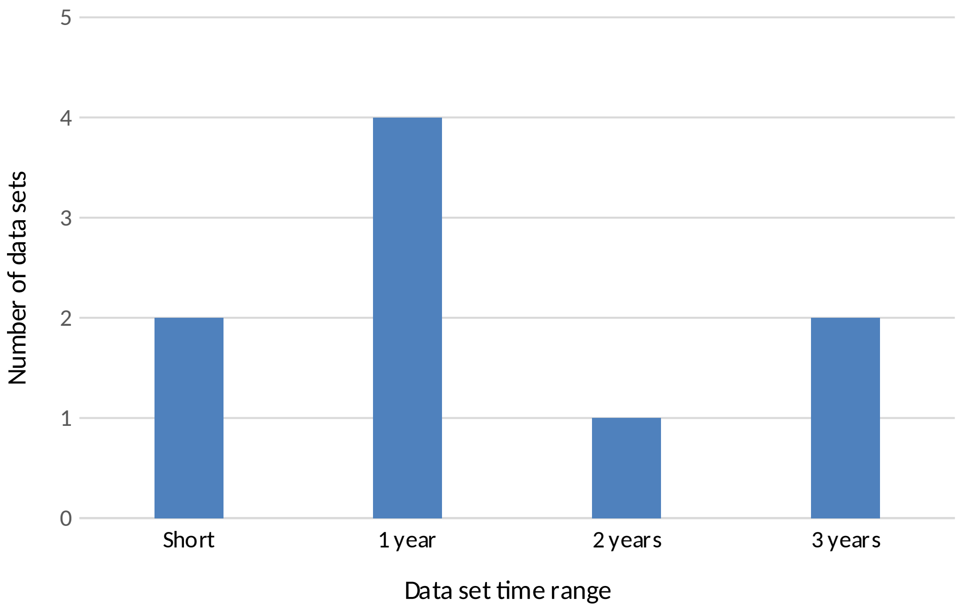

FIG. 4. Time ranges (as reported by CRP participants) of weather measurement data used for the EPZ aimed dispersion calculations. Short time refers to a manual selection of a single case.

4.3.1.1. CRP activities on near field atmospheric dispersion

Most of the CRP participants used 'traditional' or existing codes for atmospheric dispersion which where originally developed for NPP analysis. In the atmospheric dispersion modelling of releases from large NPPs, for off-site EPR purposes, typical near ranges are distances higher than 1 km up to some 20 km. For medium ranges, typical distances from tens to hundreds of kilometres are considered. This is because the site area for operating NPPs might be relatively large, even up to 1 km radius. The licensee is responsible for the site area. Furthermore, for the very near range, urgent protective actions would be required with certainty. Smaller plants with smaller footprints and/or site areas might bring a different perspective. For smaller plants, the on-site radioactive inventory is smaller, the on-site buildings and possible chimneys are smaller and the decay heat is smaller. This might reduce off-site radiological consequences from the geographical viewpoint. Not all atmospheric dispersion models are equally accurate for the very near range (i.e. for distances approximately lower than 500 m from the source of the release). There are local phenomena that cause concentration differences in the very near range, but have negligible effects for further ranges.

In addition to the range considered, the type of environment affects dispersion. Historically, existing large NPPs are most typically sited on relatively flat terrain, near cooling water, in open (not densely built) countryside. The setting is very different for urban residential areas or industrial sites where buildings increase terrain roughness, which might decrease wind speeds locally, but also induce more turbulence (building wake effects, stack tip downwash). Furthermore, at a local level releases might be significantly influenced by features of the built environment. In the simplest type of dispersion modelling (Gaussian), it is then crucial that the dispersion parameters have good validity in the very near range, but also in densely built environments. In some CRP participants' studies, the importance of densely located local weather measurements (e.g. SODAR, drones) as conditions for dispersion is mentioned.

Specifically, two participants developed their in-house codes for near field dispersion and the associated three dimensional cloud shine calculations, and another two developed CFD models for the same problem. For instance, CNL developed their 'fine-resolution atmospheric dispersion modelling platform' with as many as five options: Gaussian plume and puff, Lagrangian particle, CFD, and PGM. Tsinghua University described a complex FFT based method for three dimensional cloud shine, which implies that a near field concentration distribution can be used that is not yet dispersed horizontally much wider than gamma range in air.

The local dispersion is not described in detail. However the local dispersion was their test case TC4 a local scale simulation over heterogeneous topography with Lagrangian particles. SNERDI reported that MACCS results did not meet a 10 mSv requirement, and then resorted using CFD instead, which reduced the calculated doses and thus also the EPZ size to the site boundary to a maximum of 300 m. There is no detailed description given, but the use of CFD for ranges up to 300 m implies specific near field modelling with high resolution. Pittsburgh Technical reported that they had developed a model to estimate near field aerosol deposition from SMRs in ANSYS CFD. They performed a parametric study by varying release height, distance between reactor and another building, height of the building, wind velocity, particle mean diameter and particle density.

4.3.2. Deterministic approach

Atmospheric dispersion conditions affect the doses significantly, hence the limiting conditions should be selected carefully. In deterministic analysis, weather conditions (stability classes) reflecting the most commonly observed local conditions and/or reflecting the worst conditions ever observed locally, or numerical weather data monitored on a representative day and/or numerical weather data monitored on a particular day might be used.

Table 5 summarizes how participants in the CRP performed their studies to determine EPZ sizes for their respective SMR plants, or what kind of development they performed to further that purpose.

An EPZ study might combine deterministic and probabilistic approaches. For example, a single accident scenario might be chosen but considered with a probabilistic approach for the atmospheric dispersion of the associated source term; or several accidents might be considered, but all with a single chosen weather case (typical weather or conservative weather).

TABLE 5. OVERVIEW OF DETERMINISTIC APPROACHES ADOPTED BY CRP INSTITUTES FOR ATMOSPHERIC DISPERSION AND DEPOSITION

Participating institute	**Member State**	**Reactor name**	**Type of reactor**	**Power MW (th)/(e)**	**Outcomes of study**	**On-site scenarios selected**	**Codes used (dispersion and dose projection), DC factors**	**Weather data**	**Deterministic or probabilistic; best estimate or conservative**
CNEA	Argentina	CAREM	PWR iPWR	100 MW(th)/30 MW(e)	IRR risk, EPZ size	Station blackout (SBO)	MACCS; EPA FGR-13	Monitored in Atucha, 8760 h of data	Prob.+det. BE
CNL	Canada	—[a]	HTGR, iPWR, MSR, LFR	30 MW(th)	Phenomena, source term, dispersion, dose projection	Release fractions from literature	In-house near field (plume, puff, particle, PGM); ADDAM; ICRP	CALMET, SODAR; data monitored at Chalk River Laboratories over 3 years	Det. conserv.+prob.
CNPE (CNNC)	China	ACP100	iPWR	385 MW(th) (125 MW (e))	EPZ size	DBA; DEC-B; Level 2 PSA	—	Conserv.+realistic	Det. (many cases) +prob.
KAERI	Republic of Korea	SMART	iPWR	365 MW (th) (107 MW (e))	Sequences, source term, dispersion, dose projection, EPZ	SBO, SGTR; 5 source term categories	MELCOR, MACCS	Mast, coastal, 8760 h of data; also yearly average of 45 stations	Det.+prob.; BE+conserv.
SOREQ	Israel	NuScale	iPWR	77 MW(e)	Scenarios, source term	GPW strike	Serpent, OpenFOAM	n.a.[b]	Det.
JAEA	Japan	—	Prismatic HTGR	600 MW (th)	Scenario, sequences, source term, dispersion, dose projection, EPZ	Primary co-axial double pipes	RELAP5, in-house (THYTAN, HTFP, FORNAX-A), NUPRA, OSCAAR;	Representative (most common/average)	Prob.+det.; conserv.

TABLE 5. OVERVIEW OF DETERMINISTIC APPROACHES ADOPTED BY CRP INSTITUTES FOR ATMOSPHERIC DISPERSION AND DEPOSITION (CONT.)

Participating institute	**Member State**	**Reactor name**	**Type of reactor**	**Power MW (th)/(e)**	**Outcomes of study**	**On-site scenarios selected**	**Codes used (dispersion and dose projection), DC factors**	**Weather data**	**Deterministic or probabilistic; best estimate or conservative**
Toshiba	Japan	4S	SFR	30 MW (th)/15 MW(e)	Scenario; doses	Cladding failure without core melt	ORIGEN, Excel; EPA-520/1-88-020 and EPA-402-R-93-081 [41, 42]	—	Det.
EC/JRC	n.a.	—	n.a.	—	Sequences, ST, dispersion, dose projection; rationale	Worst case vs PSA	—	—	Det.+prob.
PAEC	Pakistan	—	iPWR	50/100 MW(e)	Source term, dispersion, dose projection, EPZ	5/10% core volatiles	MACCS	Hourly direction, wind speed, stability, rain	Det.+prob.; BE
STEG	Tunisia	ACP100	iPWR	100 MW(e)/385 MW(th)	Scenario, source term, dispersion, dose projectio, EPZ	LBLOCA, 20 nuclidesFT	ADMS, PC-CREAM, PACE	Wind direction and speed, stability	Prob.+det.; BE

[a] —: data not available.
[b] n.a.: not applicable.

4.3.3. Probabilistic approach

TABLE 6. OVERVIEW OF PROBABILISTIC APPROACHES ADOPTED BY CRP INSTITUTES FOR ATMOSPHERIC DISPERSION AND DEPOSITION

Participating institute	**Member State**	**Reactor name**	**Type of reactor**	**Power MW (th)/(e)**	**Outcomes of study**	**On-site scenarios selected**	**Codes used (dispersion and dose projection), DC factors**	**Weather data**	**Deterministic or probabilistic; best estimate (BE) or conservative**
CNEA (RCM4)	Argentina	CAREM	PWR iPWR	100 MW(th)/30 MW(e)	IRR risk, EPZ size	SBO	MACCS; EPA FGR-13	Monitored in Atucha, 8760 h of data	Prob.+ det.; BE
CNL	Canada	—[a]	HTGR, iPWR, MSR, LFR	30 MW(th)	Phenomena, source term, dispersion, dose projection	Release fractions from literature	In-house near field (plume, puff, particle, PGM); ADDAM; ICRP	CALMET, SODAR; data monitored at Chalk River Laboratories over 3 years	Det. conserv.; prob.
CNPE (CNNC)	China	ACP 100	iPWR	385 MW(th) (125 MW(e)	EPZ	DBA ; DEC-B; Level 2 PSA	—	Conserv. +realistic	Det. (many cases)+prob.
SNERDI	China	CAP 200	iPWR	100 MW(e)/660 MW(th)	—	FHA	MACCS, CFD	Mast, 3 years hourly (direction, speed, stab, rain)	Prob.
Tsinghua INET	China	HTR-PM	HTGR	210 MW(e)	Phenomena, source term, dispersion, dose projection	SGTR; NUREG-0396 [20]; P2, P4, SG61, SG91, T4	KORIGEN, FRESCO2; PAVAN, MACCS, in-house models for near field	Conserv.	Prob.

							(FFT, NFFT)		

TABLE 6. OVERVIEW OF PROBABILISTIC APPROACHES ADOPTED BY CRP INSTITUTES FOR ATMOSPHERIC DISPERSION AND DEPOSITION (CONT.)

Participating institute	**Member State**	**Reactor name**	**Type of reactor**	**Power MW (th)/(e)**	**Outcomes of study**	**On-site scenarios selected**	**Codes used (dispersion and dose projection), DC factors**	**Weather data**	**Deterministic or probabilistic; best estimate (BE) or conservative**
VTT	Finland	SMART	iPWR	365 MW(th) (107 MW(e))	Dispersion, dose projection, EPZ size	MSLB and SGTR; 57 nuclides	MACCS, ARANO, VALMA	Mast data monitored over 8760 h	Prob.; BE
KAERI	Republic of Korea	SMART	iPWR	365 MW(th) (107 MW(e)	Sequences, source term, dispersion, dose projection, EPZ	SBO, SGTR; five source term categories	MELCOR, MACCS	Mast, coastal, data monitored over 8760 h; also yearly average of 45 stations	Det.+prob.; BE +conserv.
K.A.CARE	Saudi Arabia	SMART	iPWR	365 MW(th) (107 MW(e)	Dispersion, dose projection	SBO, SGTR; five source term categories	HotSpot	Mast data, desertic area data, 2 years hourly	Prob.
BATAN/BRIN	Indonesia	—	PWR	100 MW(e)/300 MW(th)	EPZ	LBLOCA, based on large PWR	ORIGEN, COSYMA	Mast, 1 year hourly	Prob.
JAEA	Japan	—	Prismatic HTGR	600 th	Scenario, sequences, source term, dispersion, dose projection, EPZ	Primary co-axial double pipes	RELAP5, in-house (THYTAN, HTFP, FORNAX-A), NUPRA, OSCAAR;	Representative (most common/average)	Prob.+det.; conserv.

							ICRP and NRA		

TABLE 6. OVERVIEW OF PROBABILISTIC APPROACHES ADOPTED BY CRP INSTITUTES FOR ATMOSPHERIC DISPERSION AND DEPOSITION (CONT.)

Participating institute	**Member State**	**Reactor name**	**Type of reactor**	**Power MW (th)/(e)**	**Outcomes of study**	**On-site scenarios selected**	**Codes used (dispersion and dose projection), DC factors**	**Weather data**	**Deterministic or probabilistic; best estimate (BE) or conservative**
EC/JRC	n.a.[b]	—	n.a.		Sequences, source term, dispersion, dose projection	Worst case vs PSA	—	—	Det.+prob.
PAEC	Pakistan	—	iPWR	50/100 MW(e)	Source term, dispersion, dose projection, EPZ	5/10% core volatiles	MACCS	Hourly wind direction and speed, stability class, rain	Det.+prob.; BE
STEG	Tunisia	ACP100	iPWR	100 MW(e)/385 MW(th)	Scenario, source term, dispersion, dose projection, EPZ	LBLOCA, 20 nuclides	ADMS, PC-CREAM, PACE	Wind direction and speed, stability class	Prob.+det.; BE
CRA	United Kingdom	NuScale	iPWR MSR	50 MW(e)	Dispersion, dose projection, risk, EPZ, health effects, costs	LIS/SIS LOCA, 30 nuclides	PACE, NAME III	Wind direction and speed, rain	Prob.; BE

[a] —: data not available.
[b] n.a.: not applicable.

Output from full PSA (up to Level 3) can be probability distributions of off-site doses, as well as the ‘dominant sequence’ (i.e. the sequence that brings most contribution to the off-site dose/ health risks).

The necessary phases of a Level 3 PSA procedure (based on Level 2 PSA results) are briefly outlined in this section. The main input needed for EPZ size calculations is a source term or a set of source terms. Usually, only a few different representative source terms, with significant probabilities from Level 2 PSA, can be calculated.

In terms of numerical weather data, data from a three-dimensional numerical weather prediction model is usually considered to be best. However, simpler models with simpler weather data (e.g. one-point data from a mast) helps calculation codes run faster, and errors in the release plume winding trajectory will usually cancel out in a probabilistic study that considers a sufficiently long time period.

For Level 3 PSA studies, site specific long term (for five to ten years) weather data might be needed to thoroughly account for the probability distributions of the various weather parameters.

Weather mast measurements might be more reliable than numerical weather prediction modelling to acquire the dispersion parameters near the source. More masts would make the data even more reliable and representative of the local conditions.

4.4. DOSE ASSESSMENT

Table 7 provides an overview of the whole chain of dose projection expertise as applicable and adopted by the institutes participated in this CRP.

TABLE 7. OVERVIEW OF THE WHOLE CHAIN OF EXPERTISE FOLLOWED BY CRP INSTITUTES

Institute	Member State	Reactor name	Type of reactor	Power MW (th)/(e)	Steps, outcome of study	Scenarios selected, nuclides	Codes used (mainly doses and dispersion), DC factors	Weather data	Dose exposure pathways; target organs; integration time	Protection /dose limits	Deterministic or probabilistic; best estimate (BE) or conservative	National regulator
CNEA	Argentina	Surry; CAREM	PWR iPWR (SMR)	2587 30 e 300 MW(th)	IRR, EPZ size	Source term SBO	MACCS; EPA FGR-13	Data monitored at Atucha over 8760 h	Cloud shine, ground shine, inhalation	Sheltering, evacuation, relocation	Prob.+det.; BE	ARN
CNL	Canada	—[a]	HTGR, iPWR, MSR, LFR	30 MW(th)	Phenomena, source term, dispersion, dose projection	Release fractions from literature	In-house near field (plume, puff, particle, PGM); ADDAM; ICRP	CALMET, SODAR; data monitored at Chalk River Laboratories over 3 years	3-D cloud shine; cloud shine, ground shine, inhalation	Shelter, evac	Det.+prob.; conserv	CNSC
CNPE (CNNC)	China	ACP 100	iPWR	385 MW(th) (125 MW(e))	EPZ sizes	DBA; DEC-B; Level 2 PSA	—	Conserv. +realistic	Cloud shine, inhalation; thyroid, lung	GB 18871; sheltering, evacuation, ITB	Det. (many cases) +prob.	NNSA
SNERDI	China	CAP 200	iPWR	100 MW(e)/660 MW(th)	—	FHA	MACCS, CFD	Mast, 3 years hourly (wind direction and speed, stability class, rain)	—	GB 18871, GB/T 17680.1; sheltering, evacuation, ITB	Prob.	NNSA

Institute	Member State	Reactor name	Type of reactor	Power MW (th)/(e)	Steps, outcome of study	Scenarios selected, nuclides	Codes used (mainly doses and dispersion), DC factors	Weather data	Dose exposure pathways; target organs; integration time	Protection /dose limits	Deterministic or probabilistic; best estimate (BE) or conservative	National regulator
Tsinghua INET	China	HTR-PM	HTGR	210 MW(e)	phenomena, source term, dispersion, dose projection	SGTR; NUREG-0396; P2, P4, SG61, SG91, T4	KORIGEN, FRESCO2; PAVAN, MACCS, in-house models for near field (FFT, NFFT)	Conserv.	3-D cloud shine	—	Prob.; BE	NNSA
VTT	Finland	SMART	iPWR	365 MW(th) (100 MW (e))	Dispersion, dose projection, EPZ size	MSLB and SGTR; 57 nuclides	MACCS, ARANO, VALMA	Mast; data monitored over 8760 h	Cloud shine, ground shine, inhalation; red marrow, skin, thyroid, lungs, large intestine	Sheltering, evacuation (GSR Part 7 [4]; STUK)	Prob.; BE	STUK
KAERI	Republic of Korea	SMART	iPWR	365 MW(th) (107 MW(e))	Sequences, source term, dispersion, dose projection, EPZ	SBO, SGTR; STC 1–5 (UCA, LCA, CI, BP)	MELCOR, MACCS	Mast, coastal, 8760 h; also yearly average of 45 stations	Cloud shine, ground shine, inhalation; red marrow+ effective dose	Shelter, evacuation, ITB, reloc ation (GSR Part 7 [4]; NSSC)	Det.+prob.; BE, +conserv.	NSSC

Institute	Member State	Reactor name	Type of reactor	Power MW (th)/(e)	Steps, outcome of study	Scenarios selected, nuclides	Codes used (mainly doses and dispersion), DC factors	Weather data	Dose exposure pathways; target organs; integration time	Protection /dose limits	Deterministic or probabilistic; best estimate (BE) or conservative	National regulator
K.A.CARE	Saudi Arabia	SMART	iPWR	365 MW(th) (107 MW(e))	Dispersion, dose projection	SBO, SGTR; 5 source term categories	HotSpot	Mast, desertic area, 2 years hourly	Cloud shine, ground shine, inhalation; thyroid+ effective dose	—	Prob.	SAARA
BRIN	Indonesia	—	PWR	100 MW(e) /300 MW(th)	EPZ sizes	LBLOCA, based on large PWR	ORIGEN, COSYMA	Mast, 1 year hourly	Cloud shine, ground shine, inhalation, ingestion	Sheltering, evacuation, ITB, relocation, food ban, decontamination	Prob.	BAPETEN
SOREQ	Israel	NuScale	iPWR	77 MW(e)	Scenarios, source term	GPW strike	Serpent, OpenFOAM	n.a.[b]	n.a.	n.a.	Det.	NLSO

Institute	Member State	Reactor name	Type of reactor	Power MW (th)/(e)	Steps, outcome of study	Scenarios selected, nuclides	Codes used (mainly doses and dispersion), DC factors	Weather data	Dose exposure pathways; target organs; integration time	Protection /dose limits	Deterministic or probabilistic; best estimate (BE) or conservative	National regulator
JAEA	Japan		Prismatic HTGR	600 MW(th)	Scenario, sequences, source term, dispersion, dose projection, EPZ	Primary co-axial double pipes	RELAP5, in-house (THYTAN, HTFP, FORNAX-A), NUPRA, OSCAAR; ICRP and NRA	Representative (most common/average)	Cloud shine, inhalation; foetus, embryo, red marrow, thyroid	Sheltering, ITB	Prob.+det.; conserv.	NRA
Toshiba	Japan	4S	SFR	30 MW(th)/ 15 MW(e)	Scenario, dose projection	Cladding failure with or without core melt	ORIGEN, Excel; EPA-520/1-88-020 and EPA-402-R-93-081[46, 47]		Total effective dose equivalent	10 mSv	Det.	NRA
EC/JRC	n.a.		n.a.		Sequences, source terms, dispersion, dose projection; rationale	Worst case vs PSA			Cloud shine, ground shine, inhalation, ingestion; thyroid, foetus		Det. vs prob.	n.a.

Institute	Member State	Reactor name	Type of reactor	Power MW (th)/(e)	Steps, outcome of study	Scenarios selected, nuclides	Codes used (mainly doses and dispersion), DC factors	Weather data	Dose exposure pathways; target organs; integration time	Protection /dose limits	Deterministic or probabilistic; best estimate (BE) or conservative	National regulator
PAEC	Pakistan		iPWR	50/100 MW(e)	Source term, dispersion, dose projection, EPZ	5/10% core volatiles	MACCS	Hourly wind direction and speed, stability, rain	Cloud shine, ground shine, inhalation; effective 1/7 d, foetus	Evacuation, sheltering, relocation, ITB; IAEA GSR Part 7 [4] and EPR-NPP-PPA[11]	Det.+prob.; BE	PNRA
STEG	Tunisia	ACP100	iPWR	100 MW(e)/385 MW(th)	Scenario, source term, dispersion, dose projection, EPZ	LBLOCA, 20 nuclides	ADMS, PC-CREAM, PACE	Wind direction and speed, stability			Prob.+det.; BE	None
CRA	United Kingdom	NuScale	iPWR MSR	50 MW(e)	Dispersion, dose projection, risk, EPZ, health effects, costs	LOCA, 30 nuclides	PACE, NAME III	Direction and wind speed, rain		Evacuation, sheltering	Prob.; BE	ONR
Texas A&M	United States of America		MHTGR		Source term		GOTHIC, STAR-CCM+	n.a.	n.a.	n.a.		NRC

Institute	Member State	Reactor name	Type of reactor	Power MW (th)/(e)	Steps, outcome of study	Scenarios selected, nuclides	Codes used (mainly doses and dispersion), DC factors	Weather data	Dose exposure pathways; target organs; integration time	Protection /dose limits	Deterministic or probabilistic; best estimate (BE) or conservative	National regulator
18. ANL	United States of America		SFR		Source term			n.a.	n.a.	n.a.		NRC
Pittsburgh	United States of America		iPWR		Source term, dispersion, dose projection, protective actions		ANSYS, RASCAL; importance of near field	Wind speed in ANSYS parametric study		Evacuation variation tool		NRC

[a] —: data not available.
[b] n.a.: not applicable.

4.4.1. Methodology for dose projection

Most computer codes used by CRP participants are dedicated tools for off-site dose projection and they are integrated in the sense that atmospheric dispersion and doses usually come within the same package, instead of having a separate module for dispersion and then explicitly transferring the concentration and deposition results to the dose part.

Dose exposure pathways refer to the means and route by which radiation might reach the receptor. Depending on the location of the radiation source, pathways are classified into external (source outside the body) and internal (inside body). Practically, alpha and beta emitters are only effective in causing internal doses, whereas gamma photons' effective range in air is up to hundreds of metres. The external doses calculated by CRP participants were those from cloud shine (mostly gamma, directly from the radioactive plume as it is passing by) and ground shine (mostly gamma, from radionuclides deposited as fallout on the ground surface). Additionally, VTT also presented the dose to skin from nuclides deposited on the skin in MACCS calculations for the SMART reactor. For internal dose, practically only that from inhaled radionuclides was considered, although BRIN also presented that for ingested nuclides in their COSYMA calculations, apparently for determination of EPDs and ICPDs. Internal exposure causes a dose commitment; that is, the dose will accumulate from radionuclides inside the body over a long period of time. Depending on location, resuspension of deposited substances might later add to that from the inhalation of nuclides. However, no CRP participant mentioned having considered resuspension as a dose exposure pathway.

Figure 5 shows how many participants considered each of the mentioned dose pathways. Cloud shine in three dimensional detail of the radiation source is mentioned separately because of its special importance in the near field. It was considered by participants who also developed dedicated near field dispersion modelling. Of all the participants, seven explicitly reported having included all the three main dose pathways (cloud shine, ground shine, inhalation).

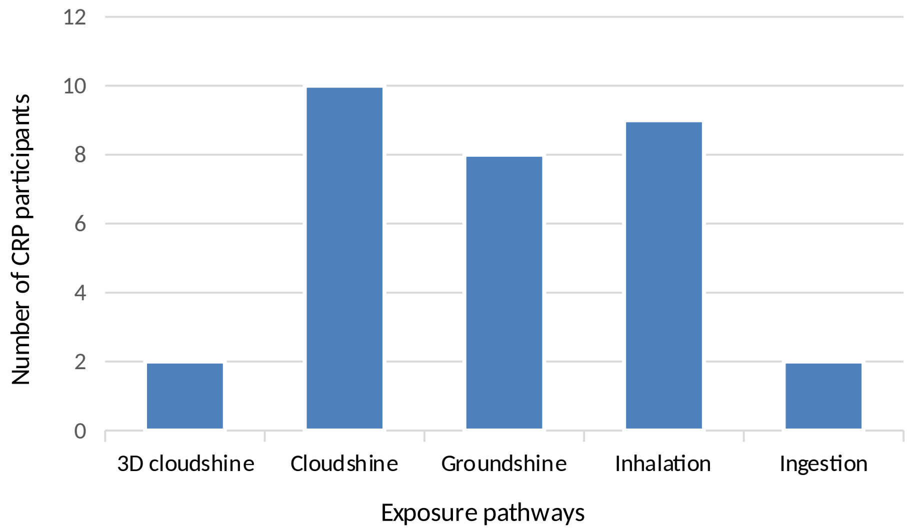

FIG. 5. Dose exposure pathways considered by CRP participants.

The main external exposure pathways that were considered in the CRP are direct exposure from the radioactive cloud or ground deposition, while internal pathways are through inhaled or ingested radionuclides.

From any of the dose pathways considered in the CRP, doses will be received all over the body, in various target organs or tissues, each having its specific dose conversion factor (DCF). Several sources for nuclide specific DCFs are available, and most participants did not specify in their report which set of DCFs they had used. The dose projection tool might have a fixed set of DCFs or offer the user the choice between several DCF set options. Appendix II of GSR Part 7 [4] mentions red bone marrow, foetus, skin, thyroid, lung and colon for deterministic effects, and thyroid and foetus for stochastic effects. Depending on the calculation code used by each CRP participant, a variable selection of target organs might be available. Most calculation codes, however, generate effective dose (weighted average of organ-specific doses).

Appendix II of GSR Part 7 [4] provides suggested generic criteria for use in EPR. Generic criteria are defined as "Levels for the projected dose, or the dose that has been received, at which protective actions and other response actions are to be taken".

Figure 6 shows individual receptor organs considered in the CRP. The most represented target organs were thyroid, red marrow and foetus, which appear to be important in terms of adverse health effects.

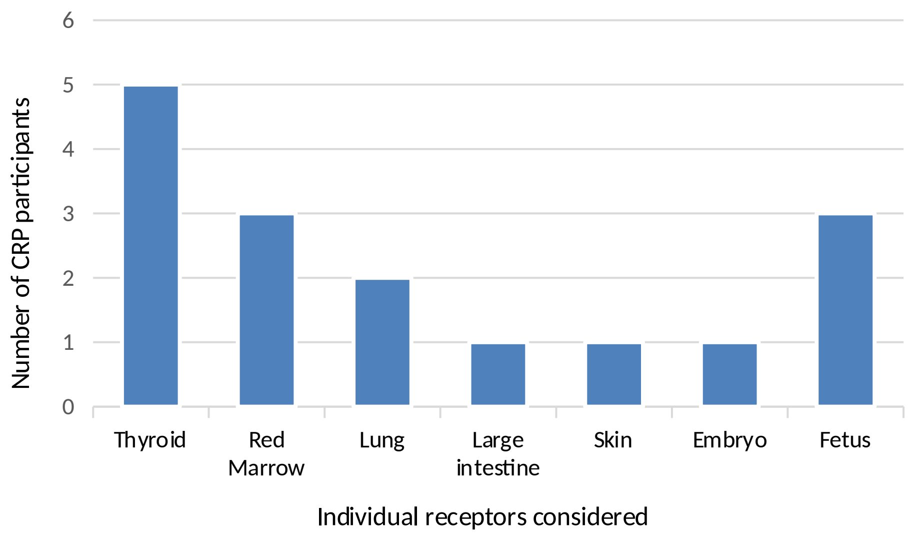

FIG. 6. Individual receptor organs considered in the CRP.

Precautionary urgent or urgent protective actions in EPZs would be ordered and implemented if generic criteria associated with such are exceeded or projected to be exceeded. Dose levels resulting from an NPP accident are expected to decrease with distance from the plant, although local dose level maxima might exist at certain distances from the plant, for example due to a high effective release height or wet deposition from rain showers along the cloud transport route.

CRP participants set the upper (outer) bound of the EPZ sizes they estimated at the largest distance from the SMR where, according to their atmospheric dispersion and dose projection simulations, generic criteria could be exceeded. Most CRP participants used the generic criteria set by their national regulator; others applied the IAEA generic criteria given in Appendix II of GSR Part 7 [4]. Figure 7 shows how many CRP participants explicitly mentioned considering each protective action in their EPZ study. The most represented protective actions were sheltering, evacuation and ITB.

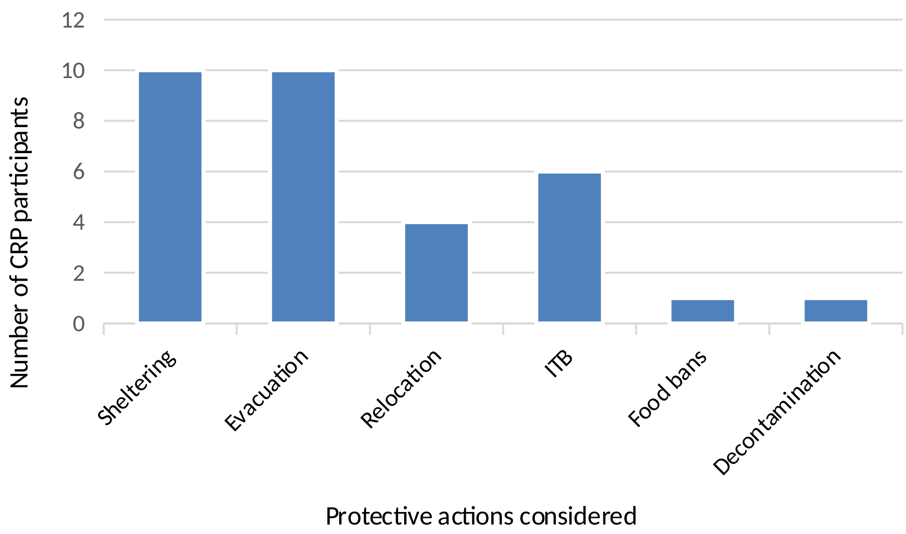

FIG. 7. Protective actions reportedly considered in CRP participants' dose calculations.

4.4.1.1. Population groups

Depending on each country's regulatory requirements and the availability of dose conversion factors (DCFs), it is possible and usual to calculate doses for certain subgroups of the whole population (e.g. 'most exposed group', or different age groups such as adults and children/infants, often subdivided into many age ranges, or according to habits/occupation type), not only the average of all. The doses to different population groups are generally different due to differences in indoors/outdoors time, physical activity, shielding, or the differences in dose conversion factors.

In the CRP, only a few participants explicitly mentioned the target groups in their studies. CNL reported that even when the ADDAM code allows DCF input for different age groups, they used the adult individual (committed effective dose over 3.5 days), consistent with the CSA N288.2 standard [48]. This is apparently also the approach used by most participants. SNERDI indicated using population data validated within the last five years. BRIN used population data taken once a year.

4.4.1.2. *Shielding factors*

Shielding factors are dimensionless coefficients that allow us to take landscape shapes, vegetation and buildings into account with respect to how they shade/attenuate the receptor person from external radiation. Usually cloud shine and ground shine are first calculated for a person standing in open air on a completely flat ground surface. But in practice, the consideration of a non-flat terrain might suggest multiplying the projected ground shine doses by a factor of 0.7 (terrain self-shielding). One family or semidetached houses might provide a shielding factor of 0.4 and apartment buildings a shielding factor of 0.1. With assumptions of housing choice distribution and residency times indoors/outdoors, a weighted average might be calculated, if needed. Similar considerations apply for cloud shine. For inhalation dose, a 'shielding factor' might be used to account for the reduction in breathing air concentrations during residence/sheltering indoors, as compared with outside breathing.

Most CRP participants did not detail their assumptions regarding shielding, and it might be that they conservatively assumed no shielding was available. CNL reported applying no shielding factors, even when not restricted by the ADDAM code. PAEC mentioned considering the shielding situation in their scenarios and the use of dose reduction factors for different pathways and structures, but did not share shielding factor values. VTT used 0.3 for ground shine, but no shielding (1.0) for other exposure pathways.

4.4.1.3. *Dose conversion factors*

Dose conversion factors (DCFs) are the practical basis for transforming concentration quantities into radiation doses. For external pathways, they relate the prevailing dose rate to the ambient unit concentration of a certain nuclide: cloud shine DCFs are in the form (Sv/s)/(Bq/m^3) and ground shine DCFs are in the form (Sv/s)/(Bq/m^2), usually defined for a person standing in open air on a completely flat infinite ground surface. Time integration of the dose rate up to a specified integration time will then provide the dose. For internal pathways, the form is simply Sv/Bq, meaning that per unit of activity intake, the person is committed to receive a certain dose over a specified period of time (e.g. the next 50 years of life), hence the term 'dose commitment' or 'committed dose'. Basically, a DCF value exists for each combination of exposure path, nuclide and target organ (or effective dose). The internal DCFs are typically also available for different combinations of age group and integration time.

In cases of exposure to radiation, each target organ/tissue has its specific DCF. Several international reference publications on nuclide specific DCFs are available. The use of one or another reference publication might lead to discrepancies in nuclide specific DCF values and, consequently, differences in dose projection results (all other technical assumptions and input data being equal). A dose projection tool might have a fixed set of DCFs or offer the user the choice between several set DCF options. As for population groups or shielding factors, most CRP participants did not specify in their reports which set of DCFs they had used. CNL mentioned that the DCF values used (and taken as input for the ADDAM code) are based on international standards such as those established by the International Commission on Radiological Protection (ICRP). Toshiba indicated that their DCFs were based on publication from the United States of America, namely EPA-520/1-88-020 and EPA-402-R-93-081 [46, 47]. JAEA referred to DCFs by the ICRP. CNEA used DCFs from EPA 402-R-99-001 [49] in MACCS.

As MACCS was the most commonly used dose projection tool in the CRP (but in different versions of the software), the available DCF choices in MACCS2 and WinMACCS are as follows:

— DOSFAC2: cloud shine and ground shine from DOE/EH-0070 [50], inhalation and ingestion ORNL (1987), 60 nuclides;
— FGRDCF: EPA Federal Guidance Reports (FGRs), 600 or 825 nuclides; current WinMACCS (manual SAND-2021-1588) using EPA 402-R-99-001 [49], 825 nuclides;
— IDCF2: cloud shine and ground shine from DOE/EH-0070 [50], inhalation and ingestion by IDCF code of INL, 500 or 800 nuclides.

4.4.2. Interpretation of dose results in deterministic and probabilistic approaches

The EPZ determination methodologies used in the CRP can be broadly divided into two categories, based on the atmospheric dispersion approach that was chosen to calculate activity concentrations, serving as input to the dose projection. That choice defines the volume of output data generated. With deterministic atmopsheric dispersion, that is, one weather case (or a few weather cases), a unique dose versus distance relation (or a few dose versus distance relations) is obtained. With probabilistic atmospheric dispersion, each dose value has a probabilistic distribution, with the number of samples being defined by the number of weather cases used (typically thousands). For both types of atmopsheric dispersion, dose versus distance results are usually obtained for several exposure pathways. Moreover, for each type of atmospheric dispersion, the selection of accidental release source terms might have been deterministic or probabilistic.

With deterministic atmospheric dispersion, CRP participants usually assessed the maximum distance from the source at which generic crtieria for protective actions would no longer be exceeded.

With probablistic atmospheric dispersion, CRP participants usually chose a fractile threshold, 95% in all probabilistic cases considered in the CRP, formulating outputs such as, “in 95% of the calculations performed, the maximum projected dose was within the interval [value 1; value 2]”. Another possibility would be to plot the probability of exceeding the dose limit (for a certain protective action) versus distance. Using both dose projections and their associated probabilities, risks can be quantified (risk being defined as probability times consequences). Furthermore, projected doses might be interpreted further to examine probabilities of experiencing stochastic health effects, using the dose–response functions for various such effects. In the CRP, CNEA used risk (IRR) as their final quantity of interest; however, CNEA only looked at fatality as a consequence.

In the majority of cases, CRP participants followed a probabilistic approach with a fractile type threshold. Usually, such an approach leads to a vast amount of numerical dose results being generated: different locations (or distances to the source), time points, internal and external exposure pathways, considering the various combinations of source terms and weather scenarios.

In a nutshell, probabilistic studies conducted as part of the CRP proceeded as follows:

— From Level 3 PSA of a probabilistic study, one obtains off-site doses (possibly also health and economic effects) and their frequencies (probability distribution, most traditionally presented in the form of a complementary cumulative density function): probability of exceeding any given value of dose.

— Pick the relevant ones from all the dose results that were output from the dose projection tool.
— Obtain relevant dose exposure pathways (e.g. inhalation, cloud shine, ground shine; bone marrow, lungs, effective dose).
— Obtain relevant exposure time periods (e.g. those appearing in the criteria for various radiological countermeasures).
— Choose fractiles (typically 95% or 99.5%, as conditional probability after one certain release already has occurred), that is, the dose level that is exceeded only with low probability when the cases are those calculated with all available atmospheric source terms and weather situations; then produce dose versus distance curves with that confidence level of dose.
— Compare projected doses with IAEA generic criteria or national criteria, expressed as doses, for implementation of protective actions and other response actions.
— Obtain, as a result, distances from the source that are representative of the possible size of the EPZ (to find the distance from the source up to which a generic criterion for protective action can be exceeded with a minimum set probability).

4.4.3. Generic criteria and protective measures

Usually the dose levels resulting from an accident are expected to decrease with distance from the source, although local maxima might happen (e.g. by high effective release height or wet deposition from rain showers along the cloud transport route). Practically, the CRP participants set the upper (outer) bound of the EPZ area at the largest distance from the SMR where, according to their dispersion and dose modelling, some of the generic criteria could be exceeded. First, it is considered that implementing protective actions if/where dose projections exceed generic criteria for protective actions will cause more good than harm (unnecessary trouble, cost, panic, etc.). Most CRP participants used the generic criteria set by their national regulators; others applied the IAEA generic criteria given in Appendix II of GSR Part 7 [4]. These two choices are mostly quite consistent, but some differences might exist.

As MACCS was the dose projection tool that was the most used by the CRP participants, Table 8 provides an example of the use of MACCS made by VTT to calculate off-site doses.

TABLE 8. DOSES CALCULATED BY VTT USING MACCS TO SIZE THE EPZ OF THE SMART REACTOR

Exposure type	Exposure duration	Organ	Dose type
External	10 h	Red marrow	Absorbed
External	10 h	Skin	Absorbed
Internal	30 days	Red marrow	Absorbed ($Z \geq 90$)
Internal	30 days	Red marrow	Absorbed ($Z \leq 89$)
Internal	30 days	Thyroid	Absorbed
Internal	30 days	Lungs	Absorbed
Internal	30 days	Colon	Absorbed
Internal+external	7 days	Thyroid	Equivalent
Internal+external	7 days	Effective dose	Effective

Internal+external	First year	Effective dose	Effective

The external exposure pathways considered were cloud shine, ground shine and deposition onto skin. The only internal exposure pathway considered was inhalation; ingestion was not considered. Rows 1–7 of Table 8 correspond to Table II.1 (deterministic health effects) of GSR Part 7 [4], and rows 8–10 to Table II.2 (stochastic health effects). (See GSR Part 7, Appendix II, Generic criteria for use in emergency preparedness and response [4]).

The criteria in GSR Part 7 Appendix II [4] do not specify which protective actions are to be taken, whereas the national EPR guidelines usually distinguish between protective actions (e.g. STUK in Finland specifies sheltering indoors using the generic criteria of 10 mSv in 2 days, and evacuation with the generic criteria of 20 mSv in 1 week).

As an example of more detailed study of a protective action, Pittsburgh Technical created a tool for variation and optimization of evacuation schemes. Doses calculated by the RASCAL dose projection tool are overlaid by the developed population movement model. The tool allows variation of critical parameters, including time between occurrence of accident, notification to evacuate, and the actual population movement.

4.4.4. Deciding on an EPZ size

Considering dose criteria for deciding on the size of an EPZ is necessary but not sufficient. Other parameters need to be taken into account as well.

4.4.4.1. Categorization

Possibilities to categorize different processes for deciding on the size of an EPZ were studied by JRC and are listred below. Furthermore, JRC analysed existing underlying criteria for the determination of EPZ sizes, and identified driving criteria in various countries (e.g. protective countermeasures, implementation time, plant specific calculations). The points listed below are not mutually exclusive (several of those points might be combined).

- Protective action driven EPZ and reference dose driven EPZ:
  - Protective action driven EPZ: as many EPZ subzones as there are different protective action dose criteria are considered (i.e. an EPZ subzone for evacuation, another for sheltering, another for ITB);
  - Reference dose driven EPZ: a single generic dose criterion is applied to EPZ sizing.
- Automatically triggered actions based EPZ and assessment-driven EPZ:
  - Automatically triggered actions after a certain emergency class is declared (see emergency classes defined in GSR Part 7 [4]);
  - Assessment driven EPZ: the implementation of protective actions relies on accident specific assessment and prognosis.
- Timing dependent EPZ and magnitude dependent EPZ:
  - Timing: focus on the implementation of precautionary urgent protective actions first and then on the implementation of urgent protective actions;
  - Magnitude: deterministic and/or stochastic health effects.
- Generic EPZ and plant specific EPZ:
  - Generic EPZ: equal EPZ sizes for several nuclear facilities, even when

differences exist between plant types and sites;

- Plant specific EPZ: plant and site information used for specific projections of off-site consequences, leading to site specific EPZ sizes.

Further, JRC worked on categorizing the EPZ decision making processes that rely on dose projection calculations such as those that were performed as part of the CRP (all followed plant specific approaches):

- — EPZ based on a bounding scenario: based on a limited set of 'foreseeable reasonable' accident scenarios, dose projection is conducted, which allows to assess the maximum distance(s) for protective actions;
- — EPZ based on frequency limitations: the EPZ is set such that its size covers the geographical area for which the integrated frequency (based on PSA level 2 and/or level 3) of exceeding generic criteria becomes acceptably low for the national regulatory body;
- — Risk based EPZ approach: risk is usually defined as consequence times probability. With a risk based EPZ approach, it is possible to limit the IRR from an NPP to be lower than the average public risk from other 'routine' sources of radiation exposure;
- — Risk informed EPZ approach: this approach relies on two different criteria. First, generic criteria to avoid or to minimize deterministic health effects from reasonably foreseeable accidents are set. Second, effective EPR arrangements are implemented to ensure the ability to significantly mitigate the consequences of any accidents, even those with very low frequency and high consequences.

As an example of risk based approach, CNEA performed a case study with a risk based figure of merit for the determination of EPZ distances (sizes). Distances from the source could be found from isorisk maps based on IRR. There is no deterministic fixed dose threshold, but a frequency threshold that combines off-site consequences from dose exposure and frequency of occurrence. The frequency comes from accident frequencies used in PSA. Thus the acceptance criterion makes this approach risk based. It is clearly a probabilistic approach, as PSA is needed starting from the selection of accidents.

4.4.4.2. Interpretation, observing site specific considerations in dispersion and doses

In the EPZ decision making phase, while interpreting the outcome of the dose projection with regard to EPZ sizing, it is important to know and consider the assumptions that have been made. Among the numerous assumptions are the various site specific data. Most CRP participants chose to be site specific (either the actual site for construction of the SMR, or a hypothetical but nevertheless specified location). Then it is important to understand that the results are not necessarily readily applicable to other siting options. There are many ways in which the site specific environmental conditions at an SMR site might affect the EPZ determination, for example:

- — External hazards, either natural (e.g. earthquakes, floods, sandstorms) or human induced (e.g. industrial hazards due to nearby activities, security threats). Some of these might actually be very difficult to quantify/consider in PSA methods.
- — Dispersion of radioactive nuclides is affected by the local weather conditions, such as prevailing winds, atmospheric stability and occurrence of rain.
- — Depending on soil characteristics and rain, deposition might be resuspended into the

air by, for example, high winds or sandstorms, or migrate deeper into the soil, which is generally the case in rainy/snowy regions.
— Doses are affected by local habits (people staying more indoors or outdoors, airtightness and shielding factors of the buildings, or consumption of locally produced foodstuff).
— Effectiveness of public protective actions depends on factors such as population centres, traffic connections, the presence of hospitals/elderly homes/schools, and even communications efficiency and people's educational levels.

4.4.4.3. Considerations other than doses, frequencies and distance to the source

There was no national regulatory body among the CRP participants. Assessing whether the technical results obtained constitute a solid enough basis for making a decision remains at the discretion of national regulatory bodies. Further, considerations additional to technical results obtained from dose projections need to be taken. For example, SNERDI indicated that the determination of an EPZ would be adjusted considering aspects such as the evacuation time factor (terrain, administrative division, transport, communication), evacuation costs (socioeconomic aspects) and public acceptance (to evacuate).

4.5. IDENTIFICATION OF MAJOR SOURCES OF UNCERTAINTIES

Uncertainties are inherent to each step of the methodology for estimating EPZ sizes, regardless of the approach followed (deterministic or probabilistic). These sources of uncertainties have different levels of impact on the overall results. Table 9 provides an overview of the main sources of uncertainties (list non-exhaustive) present in the main steps for EPZ size estimation.

TABLE 9. MAIN SOURCES OF UNCERTAINTIES FOR EPZ SIZING

Uncertainty source	**Uncertainty type**	**EPZ methodology step**
Accident selection	Aleatory	Accident selection approach
Source term	Epistemic	Source term calculation
Weather data	Epistemic	Radionuclide dispersion
Site characteristics	Epistemic	Site characterization
Atmospheric dispersion	Epistemic	Radionuclide dispersion
Dose calculation	Epistemic	Radiological consequences

Some of the uncertainty sources, such as uncertainties on source term calculation, cannot currently be quantified according to the corresponding state of the art. However, knowledge on other uncertainty sources is more mature, for example concerning uncertainties arising from atmospheric dispersion calculation. In addition, some of the uncertainty sources are embedded in one another, such as atmospheric dispersion model outputs that are inputs for dose projection models, which makes the whole system become complex in terms of uncertainty propagation.

4.6. FINDINGS FROM PARTICIPATING INSTITUTES ON EPZs FOR SMRs

4.6.1. Findings from the National Atomic Energy Commission, Argentina

A hypothetical accident scenario was postulated and different protection strategies were evaluated in order to study the behaviour of the IRR under the implementation of protective actions. All of the input data for the analysis were taken from reliable literature and the atmospheric dispersion and dose calculations were made with the off-site consequence assessment code MACCS.

Iso-risk curves have been obtained and plotted for the vicinity of the NPP. These curves take into account the stochastic and plant specific nature of the meteorological conditions at the release time, and the source term frequency.

As a main conclusion, CNEA considers the IRR definition proved to be particularly appropriate as a first approach for EPZ sizing. This is because its calculation considers not only the dose, through the dose–response relationship, but also takes into account the individual exposure frequency. The latter is an output of the Level 1, 2 and 3 PSA. Nevertheless, a revision or update of the IRR definition should be considered due to both the exposure time being limited to 24 h, as defined by the national licensing criteria, and the lack of considerations regarding countermeasure implementation in the IRR calculation. Accordingly, for the IRR calculations performed in the CRP, CNEA used a preliminary revision of the IRR: the exposure time was extended to one week and the application of countermeasures was introduced in the calculations.

Additionally, CNEA highlights that the definition of the EPZ size in terms of IRR is still not clear. In this sense, CNEA explored some possible IRR reference levels to define EPZ size. For instance, the results showed that individuals could be exposed to doses that exceed the thresholds for deterministic health effects inside the EPZ limited by the IRR. This is for scenarios under adverse circumstances with low probability of occurrence.

Furthermore, CNEA expressed the necessity of continuing research in this direction, and exploring the possibility of extending the definition of IRR in such a way that it could be adapted to the requirements of an emergency plan.

4.6.2. Findings from the Canadian Nuclear Laboratories, Canada

The CNL pilot study on hypothetical limiting accidents in SMRs showed that it was conceivable for the dose consequences of these accidents to be limited compared to events at larger contemporary water cooled reactors. The pilot study did not represent a complete methodology for determining EPZs in Canada because no comprehensive plant safety analyses were available to inform these determinations. Nevertheless, the results were compelling evidence that EPZs could be smaller than defined for the existing Canadian NPPs.

The predicted dose consequences in some cases in the pilot study did not exceed the lower threshold for deterministic health effects or the generic criteria for protective actions, such as sheltering beyond distances that might be considered as the site boundary. This implies that precautionary actions to limit severe deterministic health effects and other urgent protective actions off-site might not be necessary. In Canada, such actions would be defined within the automatic action zone (analogous to the PAZ) and the detailed planning zone (analogous to the UPZ), respectively. CNL highlighted that the Canadian standard CSA N1600 General Requirements for Nuclear Emergency Management Programs [51] states that a nuclear emergency response plan will reference an automatic action zone and detailed planning zone, but does not define their extent, or necessarily require that either extends beyond the site boundary.

However, the CNL pilot study also illuminated how sensitive these results could be to the assumptions for the accident source terms. Past EPZ determinations have relied on accidents identified in the plant deterministic and probabilistic safety analyses. From the CNL perspective:

- Until such detailed analyses are completed for the SMR technologies in Canada, the needlessness of PAZs and UPZs is mere supposition;
- Considering that the province is the authority having jurisdiction to define EPZ size, it is not clear how amenable the provincial stakeholders will be to this departure from the contemporary practices around large NPPs.

This is especially true for the new reactor technologies under the umbrella of SMRs with which there is little experience in Canada.

The general conclusion reached by CNL is that, whereas there is no regulatory barrier to establishing EPZs around SMRs commensurate to risk in Canada, up to the elimination of the off-site PAZ and UPZ if warranted, there is uncertainty concerning how events and accident sequences will be chosen to be included or excluded in the EPR planning basis with the consent of all stakeholders that are involved in the preparation of provincial emergency response plans. This conclusion necessarily also applies to the definition of extended planning distances because the principle of defence in depth still requires at least some consideration of off-site EPR, even if PAZs and UPZs are deemed unnecessary.

CNL’s related research on accident phenomenology and the development of new dispersion and dose modelling techniques has advanced the predictive capability of accident source terms and the determination of accident consequences. These advances might eventually impact on the determination of EPZs around SMRs in Canada as well.

4.6.3. Findings from China Nuclear Power Engineering Corporation, China

The work by CNPE was based on the 125 MW(e) ACP100 and considered 10^{-7} to 10^{-8} per reactor-year as ‘temporary’ cut-off frequency for its selection of accident sequences (and associated source terms in preliminary design) as part of its probabilistic approach. The initial ‘guess’ expressed by CNPE was that the suggested EPZ size would be smaller than 3 km.

Several types of accidents were considered by CNPE, for which China’s regulation for intervention for sheltering, evacuation and ITB was applied.

The demonstration project of Linglong One NPP with ACP100 located in Hainan Province of China was chosen as case study. First considering DBAs with larger consequences, CNPE concluded that the distance to have a maximum projected dose off-site that was lower than China’s intervention levels in terms of effective dose and thyroid dose was 300 m.

In addition, CNPE considered the representative DEC-B condition with large radiological conscequence and Level 2 PSA results for ACP100. The conclusions obtained were that:

- — With realistic atmospheric conditions, for representative DEC-B conditions, the distance to have a maximum projected dose off-site that was lower than China's intervention levels was 300 m;
- — The probability of the dose exceeding China's intervention levels at 300 m, based on all severe accident sequences of the Level 2 PSA, is far below 30%. From a technical perspective, the size of the EPZ for Hainan site would be suggested to be 300 m.

4.6.4. Findings from Shanghai Nuclear Engineering Research and Design Institute, China

Research at SNERDI used the CAP200 reactor as a basis for investigating SMR EPZ requirements and for performing EPZ case studies in China. The research findings can be summarized in the categories of: accident cut-off probability and source term, atmospheric diffusion model, identification of acceptance criteria and the final CAP200 EPZ case studies.

The draft publication Notice of the National Nuclear Accident Emergency Office on Printing and Distributing the Guiding Opinions on Land-based Small PWR Nuclear Emergency Response endorsed that 10^{-7} to 10^{-8} per reactor-year might temporarily be used as a cut-off frequency for severe accidents that determine the size of the EPZ in China. On that basis, a core damage sequence for the CAP200 reactor could not be screened out, although the containment was assumed to remain intact.

Calculations with the MACCS computer code showed that the dose projections were highly sensitive to the dispersion parameters. A CFD model for atmospheric dispersion, MSDA-3D, was therefore developed and verified against data collected in wind tunnel experiments. The doses predicted by the bespoke CFD model for CAP200 were less than half of those predicted by MACCS.

Guidelines for NPP emergency planning and preparation were reviewed in order to establish acceptance criteria for SMRs. The following requirements for SMRs were identified by SNERDI: for infrequent accidents, the effective dose at the site boundary is limited to 5 mSv within 30 days; for limiting accidents and severe accidents, the effective dose at the site boundary is limited to 10 mSv within 30 days.

Case studies on the EPZ size for the CAP200 reactor were subsequently performed. The case studies examined three sites, including one coastal site (Shidaowan) and two inland sites (Tauhuajiang and Pengze). The results showed that the probability of the dose exceeding 10 mSv at 500 m were 0.54%, 0.76% and 0.06% for the three sites, respectively. These probabilities were much smaller than the 30% exceedance probability at 10 miles suggested in [20] that SNERDI considered for comparison.

Based on these research results, a complete methodology to determine EPZ for the CAP200 SMR in China is available. SNERDI expressed its results with an EPZ that would be confined to the on-site area (<500 m).

4.6.5. Findings from Tsinghua University — Institute of Nuclear and New Energy Technology, China

The research performed by INET at Tsinghua University regarded the definition of an EPZ for the HTR-PM reactor in China. The HTR-PM is a small modular HTGR. The different research findings can be categorized as:

- Establishing the technical basis for small modular HTGR EPZs;
- Establishing the acceptance criteria for small modular HTGR EPZs in China;
- Demonstrating the feasibility of a simplified off-site response for small modular HTGRs.

Methods for determining accident sequences and evaluating accident source terms for the HTR-PM were explored using Level 2 PSA results as a basis. Five enveloping release categories for the HTR-PM were subsequently selected for nuclear emergency analysis. Ultra-fast methods for calculating three-dimensional dose rates from arbitrary radionuclide distributions were also proposed. These methods outperformed tabulated methods and semi-infinite cloud base methods for dose evaluation in validation against field measurements.

Reference [20] and China's standard GB/T 17680.1-2008 were reviewed for applicability to the HTR-PM. Discussion also took place with China's National Nuclear Safety Administration. An emergency basis release category was subsequently proposed for HTGR source term calculations.

The proposed methodology was then applied to determine the EPZ for a small modular HTGR at four Chinese nuclear sites, including the Shidao Bay Nuclear Power Plant site (the location of HTR-PM). The necessary EPZ boundary was determined by INET to be within 250 m in all cases, which would be within the site boundary. As such, off-site emergency response arrangements for HTR-PM that are less stringent than for large NPPs would be justifiable.

4.6.6. Findings from Technical Research Centre, Finland

VTT worked together with KAERI and K.A.CARE to determine an EPZ size for a SMART (100 MW(e)) nuclear power plant. VTT's role was mainly to study and propose a methodology, providing ARANO and MACCS calculations based on the accident scenario 'MSL break and SGTR one tube break' and using Olkiluoto NPP weather data. KAERI performed more extensive calculations using 'criterion c' in [44] with several source terms, aggregating the conditional probabilities of exceeding the life threatening dose limit (set at 2 Sv) from several source term categories. K.A.CARE worked on identifying site candidates in Saudi Arabia.

The 'MSL break and SGTR one tube break' SMART scenario leads to a relatively high radioactive release, with a total of 2.97 10^{15} Bq of ^{137}Cs, as compared with the severe accident source term limit set at 10^{14} Bq in Finland's regulatory guidance. Therefore, the distances at which the generic dose criteria for protective actions are exceeded are quite large (>10 km from the source). Refined distance values will depend on dose exposure pathways and choice of percentile. It should be noted that the probability of occurrence of the 'MSL break and

SGTR one tube break' SMART scenario is very low (as per the Level 2 PSA performed by KAERI).

Tentative EPZ distances (sizes; usually not a circular area in practice) for a given dose criterion (i.e. the dose criterion associated with a protective action) and probability level are shown in Fig. 8. Dimensionless distance (obtained by dividing distance results by an arbitrary fixed distance) is shown to avoid conclusions from a single source term category. Each distance entry in the table was picked conservatively as the upper bound of the distance interval obtained in MACCS. The columns of the table in Fig. 8 correspond to generic criteria found in GSR Part 7, Appendix II [4]. Each row in the figure corresponds to MACCS dose versus distance result for the shown percentile. The figure shows that a wide spectrum of tentative EPZ distances is acquired from the calculation outputs, depending on the generic criterion and percentile. The entry '-1' means that the result was missing because of how the calculation distances were set in MACCS.

		Generic Criteria in IAEA GSR Part 7, Appendix II								
		1	**2**	**4**	**5**	**6**	**7**	**8**	**9**	**10**
MACCS Dose versus Distance for the Percentile	**Mean**	3	44	9	133	3	3	722	244	289
	50%	3	22	3	89	3	3	611	178	222
	90%	3	67	22	244	9	3	944	400	494
	95%	3	89	22	289	9	3	-1	494	611
	99%	3	111	44	356	22	9	-1	611	722
	99.5%	9	111	44	378	22	9	-1	611	722
	Peak	9	133	44	494	22	9	-1	722	833

FIG. 8. Tentative EPZ distances in metre for a given dose criterion and probability level. (1. External exposure, 10 h, red marrow; 2. External exposure, 10 h, skin; 4. Internal exposure, 30 days, red marrow; 5. Internal exposure, 30 days, thyroid; 6. Internal exposure, 30 days, lungs; 7. Internal exposure, 30 days, colon; 8. Total dose, 7 days, thyroid, 9. Total effective dose, 7 days; 10. Total effective dose, 1 year).

Conclusions reached by VTT include the fact that more accurate_core inventories, conservatively chosen assumptions/calculations of released fractions, and Level 3 PSA studies of off-site consequences using real weather data retrieved over several years would help to further refine the technical outputs to inform EPZ sizes for SMRs.

4.6.7. Findings from Korea Atomic Energy Research Institute, Republic of Korea

Even though the regulatory guidelines for SMRs are not yet established in the Republic of Korea, KAERI's view is that current analysis methodology and preliminary results seem to be applicable to EPZ development. After the regulatory guidelines for SMRs have been established, it will be more straightforward to decide on EPZ sizes.

In a deterministic approach, as typical severe accident sequences for SMART, the station blackout (SBO) and SGTR sequences were selected, and the dose to distance analysis was performed for a site located in the Republic of Korea. Additionally, the deterministic dose to distance analysis for EPZ was performed for the representative accident sequences for each STC in Level 2 PSA separately. The results for the representative STC sequences are also used in the probabilistic approach to determine the EPZ size.

For the SGTR accident sequence, the results of dose projection were given, depending on plume trajectories and target organs based on the IAEA generic dose criteria.

Weather conditions monitored over one year (hourly based) were considered and dose results were expressed in the form a statistical distribution, that is, 50%, 90%, 95%, 99% and 99.5% values. The percentile value to be chosen would eventually depend on the Republic of Korea's regulatory guidelines to determine the EPZ. Generally, the higher the percentile, the larger the EPZ size.

For the SBO sequence, the PAZ and UPZ sizes are much shorter; only a few hundred metres (~200 m) would be sufficient for the UPZ.

The SGTR sequences led to much larger EPZ sizes, compared to the dose to distance in the SBO sequence. In the SGTR sequence and based on the 50% percentile, the PAZ size would be 2 km and the UPZ size would be 10 km. The basis for this is that, in the SGTR sequence, a much larger amount of fission products is assumed to be released to the environment.

Based on the Level 2 PSA source terms for the representative accident sequences, the dose to distance for the different source term categories in the Level 2 PSA was calculated. The IAEA generic criteria were again used to estimate the PAZ and UPZ sizes.

For STC 1, in which containment remains intact, the source terms are so small that a small size of EPZ was estimated. For STC 2 and STC 3, for which the containment is failed due to over pressurization of the upper containment area (UCA) and lower containment area (LCA), respectively, the source terms are much larger than for STC 1 and much longer EPZ sizes are obtained. In STC 3, when LCA is failed, UCA is assumed to fail simultaneously due to over pressurization, so the source terms of STC 3 are much larger and the associated results for EPZ size are higher than for STC 2. STC 4 corresponds to the case of the containment isolation failure in UCA, for which the containment isolation failure is similar to the case of STC 2, but the release of radioactive materials is assumed to start much earlier, so the total release fraction is much larger for STC 4. However, the possibility of the containment isolation failure of STC 4 is neglected in this study, based on SMART Level 2 PSA results. STC 5 corresponds to the case of a containment bypass sequence, and the representative accident is SGTR. However, in STC 5, no reactor vessel failure is considered, and fission products are assumed to be released to the environment through the steam generator tube rupture and the bypass of containment.

For STC 1, the PAZ and UPZ sizes obtained are few hundred metres. However, for the case of the containment failure, much longer distances are required. For the case of the containment bypass sequence (i.e. STC 5), sizes of 0.5 km for the PAZ and 20 km for the UPZ were obtained, based on 50% percentile value.

Currently, the SMART Level 2 PSA used a simple approach for a containment event tree (CET) to classify the source term categories. Moreover, the representative accident sequence for STCs was selected to be the most conservative and not the most likely. KAERI's view is that CET and source term categorization should be refined before reaching final conclusions on EPZ sizes.

In the probabilistic approach, a risk aggregation technique was used. The probabilistic approach in this study was based on [20] and follows the NEI and NuScale methodology, together with NUREG-1.242 [52] for the classification of accident sequences and acceptance criteria according to dose level.

On this basis, STC 1 was assigned to the less severe accident sequence for SMART and the conditional probability of 1 rem (10 mSv) exceedance versus distance curve shows that the EPZ size would be ~200 m.

For 'more severe accident' sequences, STC 2, STC 3, STC 4 and STC 5 are considered to integrate the results with the aggregation method. Based on the conditional probabilities (given 'more severe accidents') of dose exceeding 2 Sv whole body acute for each of the three sequences (STC 4 was neglected since its frequency fraction is zero), whose corresponding distances ranged from 25 m to 5 km, the distance at which probability drops below 10^{-3} is ~4 km.

When considering external events such as off-site fire, flood and earthquake, the containment failure frequencies for internal and external events in the SMART Level 1 PSA were used. For external events such as fire and flood, the relative frequencies of containment failure for each STC are different, which has an impact on the aggregated conditional probabilities.

4.6.8. Findings from King Abdullah City for Atomic and Renewable Energy, Saudi Arabia

Weather data monitored for a site in Saudi Arabia were used to conduct a feasibility study for construction of a SMART reactor in Saudi Arabia. The site considered shows typical desert climate conditions, reflecting Saudi Arabia's weather and topographical features. A weather station installed in 2019 at the site collected data including wind speed, wind direction and temperature at different heights. The data were collected every second from 1 October 2019 to 1 October 2021. The data were averaged hourly to prepare meteorological input files for the use of MACCS and HOTSPOT calculation codes.

Using the same source terms for SGTR and SBO sequences, and the same Level 2 PSA STCs as were used by KAERI, dose projection around Saudi Arabia's site was calculated using MACCS. The results obtained were close to the results obtained by KAERI (differences in weather conditions between sites in Saudia Arabia and the Republic of Korea did not lead to a significant difference in the dose projection results, especially when expressed as statistical distribution). K.A.CARE and KAERI consider that using numerical weather data collected over a longer time period (e.g. three or five years) would help refine the effects of local weather conditions on the dose. In addition to using MACCS, dose projection was also conducted using HOTSPOT for the purpose of comparing results. For the SGTR sequence, the generic criteria considered for protective actions (1 Gy equivalent to thyroid and lungs) led to an implementation distance of almost 2 km. Moreover, the generic criteria for protective actions expressed as total effective dose (TED) (100 mSv) led to an implementation distance of approximately 14 km. For the SBO sequence, the same two generic criteria led to an implementation distance of 0.03 km. Furthermore, the same probabilistic approach as for KAERI's research was applied. For STC 1, the EPZ size obtained was ~600 m, which is slightly larger than for the site considered by KAERI in the Republic of Korea. For 'more severe accident' cases, the conditional probability of exceeding 2 Sv around Saudi Arabia's site was the same as that for the site in the Republic of Korea and led to an EPZ size of 4 km, which was also the same as that obtained by KAERI for the site in the Republic of Korea.

4.6.9. Findings from National Research and Innovation Agency, Indonesia

The work conducted by BRIN focused on a large break LOCA scenario at a 100 MW(e) water cooled SMR. BRIN used a Gaussian model for its atmospheric dispersion simulations. The segmented Gaussian plume model is considered to be accurate by BRIN at a geographical distance 0–50 km from the source of the release. As captured in [53]:

> "[T]he Gaussian model requires the following input data and parameters: time and location, atmospheric data (wind speed and wind direction, air temperature, relative humidity, cloud cover, mixing layer height), stability class of the atmosphere (proposed by the model based on meteorological data), roughness length and information about the release (air pollutant, duration of release, amount of released material, source height)".

The meteorological data were obtained from the National Meteorrology, Climatology and Geophysics Agency. The distribution of Pasquille Guifford atmospheric stability classes as an annual base of the Serpong site were processed hourly. Radionuclides were assumed to be released through the ventilation stack (height: 60 m).

The PC Cosyma code was used to estimate radiological doses. Population distribution data and farm livestock production data in site specific areas in Indonesia (Muria Peninshula, Serang Coastal area, West Bangka and South Bangka) were obtained from Indonesia's National Statistics Agency. Doses were calculated by area within 20 km around the site for 14 distances (0.1 km, 0.25 km, 0.5 km, 0.75 km, 1.0 km, 1.5 km, 2.0 km, 2.5 km, 3.5 km, 4 km, 4.5 km, 5 km, 10 km, 20 km) and 16 direction sectors at different wind speeds.

The dose projection results obtained by BRIN were below the dose criteria for establishing EPZs.

However, BRIN concluded on the following preliminary estimations:

— PAZ with a size between 500 m and 3 km;
— UPZ with a size between 3 km and 5 km;
— EPD with a radius of 5 km;
— ICPD with a radius between 5 km and 10 km.

4.6.10. Findings from Soreq Nuclear Research Centre, Israel

The work conducted by SOREQ focused on an on-site hit by a GPW with a charge equivalent to 100 kg of trinitrotoluene. Possible scenarios can be considered based on the explosion location relative to a specific SMR module, as well as the pool's floor and surface. The most destructive case is that of a 'contact explosion' when a weapon detonates hitting or nearly ‘touching’ a module. This scenario will result in a rupture of both the reactor pressure vessel and the containment vessel and release of radionuclides. Further, a ‘less severe’ case scenario is when the explosion occurs in close proximity to a submerged object, so that it is hit both by a blast wave and by an expanding gas bubble. The blast wave will rupture the external pressure vessel and is likely to damage the internal containment vessel. The latter can be caused if not by a water blast wave, then by a high pressure, high temperature bubble of gaseous explosion products. Release of radionuclides is also a likely outcome of this scenario. In the case of a more remote explosion, a submerged object will only be hit by the blast wave, which in turn becomes weaker with distance. When such a (weakened) blast wave hits a module, even though there might not be a rupture of the internal containment vessel, the shock might damage its content. If the explosion occurs close enough to the water surface, so that the gas bubble reaches the surface and bursts into air, much of the explosion energy will also dissipate into air.

Assuming that one module was severely damaged, SOREQ applied their previously developed methodology to illustrate the evaluation of the source term created as a result of the ejection of one particular nuclide — ^{131}I. SOREQ assumed that this module is at the end of its refuelling cycle (two years). SOREQ first estimated the amount of burnt fuel: considering an SMR thermal power of 160 MWt and the energetic capacity of ^{235}U (i.e. 8.2 10^7 MJ/kg), the amount of uranium burnt in two years is ~123 kg. The ^{131}I content in this amount of fission products was estimated to be 3.5 kg. The resulting source term can be used as an initial condition for an atmospheric dispersion model to determine EPZ sizes. Such an analysis can already be performed within the current state of the methodology, bit it is likely to be rather pessimistic.

One more scenario to be considered is when a larger GPW hits an NPP building, goes through the water pool, penetrates the floor and explodes deeper in the ground underneath the building (several metres deep). GPWs capable of this are usually larger (both in terms of weight and of explosive charge) and their delivery is more complicated. In this case one can rely on some conclusions of the analysis of the effect of an earthquake on an NPP. An outcome of such an event will be pool water loss through the penetration hole in the floor and possible cracks in the floor/walls.

If some of the reactor modules are damaged and there is a release of radionuclides, an additional hazard to be considered in this case is possible contamination of the underground water. Factors such as the geological structure and the water source for the local population have to be taken into account. Regarding the possibility of burying the whole facility deeper underground, the cost would be high (construction and maintenance), for a degree of extra protection not assessed to be significant, as GPWs are capable of penetrating a few metres of soil and NPP type concrete taken together.

4.6.11. Findings from Japan Atomic Energy Agency, Japan

The work conducted by JAEA consisted of developing deterministic and probabilistic approaches for EPZ size calculations separately. In the deterministic approach, source term, atmospheric dispersion and dose projection were performed using previously identified release fractions, representative meteorological conditions, dose conversion factors and reduction factors for implemented protective actions, considering emergency scenarios at a high temperature engineering test reactor (HTTR).

In the probabilistic approach, source term, atmospheric dispersion and dose projection were performed using previously identified release fractions. Moreover, JAEA developed a Level 3 PSA code (OSCAAR), used together with WinNUPRA for the analysis of probabilistic event sequence frequencies.

For EPZ size calculations, JAEA presented dose projection results for distances ranging from 100 m to 10 km in the downwind area. The averaged relative biological effectiveness (RBE) weighted dose to red marrow and to foetus thyroid from inhalation were at least one order of magnitude smaller than the generic criteria for taking off-site protective actions to avoid or minimize deterministic effects. In addition, the results of the committed effective dose from inhalation exceeded the generic criteria for taking off-site protective actions to reduce the risk of stochastic effects, from 100 m to 155 m in the downwind distance. Furthermore, the results of the equivalent dose to the foetus following inhalation by a pregnant woman were lower than the generic criteria over all downwind distance, although a peak was observed at a distance of approximately 1 km from the site due to the contribution of fission product release from the high stack.

The conclusion reached by JAEA is that the minimum size would be below 100 m for the PAZ and 155 m for the UPZ. Those sizes would be included in the site area of an HTTR.

4.6.12. Findings from Toshiba Corporation, Japan

Toshiba reviewed the existing basis for EPZ sizing in Japan. Emergency scenarios for EPZ sizing were considered based on mitigated Class 9 event [20] without core melt due to inherent safety features of the 4S design. Toshiba calculated TEDE using Microsoft Excel and dose conversion factors based on publications from the United States of America documents [46, 47]. Results obtained in terms of relative concentration were seen by Toshiba as compliant with NUREG-1.3 and NUREG-1.4 [54, 55]. The dose criteria used as basis for estimating an EPZ size was 10 mSv TEDE, which led Toshiba to conclude on a tentative EPZ radius of 200 m for the 4S design.

4.6.13. Findings from Joint Research Centre, European Commission

The size of EPZs are bounded by distances whose determination depends on the methods, calculation assumptions and acceptance criteria set forth in national regulatory dispositions, which differ from one country to another.

Despite the lack of a harmonized set of EPZ sizes among countries, the EPZ sizes of operating and decommissioned/under decommission NPPs do not differ substantially among them; to some extent EPZ sizes ‘converged’ for most of the existing NPPs.

However, SMRs bring two challenging aspects that might prevent us from having one set of ‘converging’ EPZ sizes among countries:

— A broad power range, spanning from 20–300 MW(e) (1400% relative range), makes a single EPZ distance unsuitable for the whole range, since radiological releases are

highly dependent on the plant and reactor power. Including microreactors in this consideration extends the power range down to 1 or 2 MW(e), whereas existing reactors are usually ~500–1000 MW(e).
— SMRs do not differ solely in terms of power but also in significant aspects of design and safety, leading to qualitative and quantitative differences regarding the features of potential radiological releases (including the magnitude and isotopic composition of the release, accident kinetics, release height).

Many different approaches exist at the national and international level to determine the EPZ distances depending on the informing criteria. This section presents the underlying rationale supporting such approaches in an attempt to promote discussion and clarification of the consequences of each of them. The highlighted differences do not aim to present the spectrum of existing alternatives for EPZ determination (e.g. in computing dose or identifying a representative sequence to compute the source term), but to present the different basic criteria applied to design/define the EPZ.

EPZ sizes established on the basis of an assessment of the potential radiological consequences might follow two alternatives:

— *A protective-action driven* approach, where each EPZ is based on a specific generic criterion value that, if reached, would activate the implementation of the associated protective action;
— *A reference dose driven* approach, where the EPZ results from a single generic dose threshold application.

Accounting for as many EPZ subzones as different existing protective actions are considered, each featuring a different dose threshold, can help orient efforts in a more justified fashion. However, such further distinctions might also increase the degree of complexity in deploying the emergency response.

4.6.13.1. Protective action driven EPZ

This approach directly connects and considers as many EPZs as different protective action dose criteria are considered. Therefore, the approach leads to having, for example, one EPZ for evacuation purposes, and another for sheltering in place purposes.

4.6.13.2. Reference dose driven EPZ

This approach considers a single generic dose threshold for EPZ sizing. This is the case, for instance, for the DBA scenarios in the United States of America, where 0.05 Sv and a 0.25 Sv values for the whole body and thyroid, respectively, are used to compute the EPZ distance [20].

4.6.13.3. Automatically triggered versus assessment driven

National EPZ definitions might also be rooted depending on whether the deployment of protective actions is triggered after a certain emergency level is reached, or whether such deployment relies on an accident specific assessment of possible consequences and the prognosis of potential progression of the emergency.

Based on the latest international guidance, the deployment of protective actions to protect the public should be justified, meaning that protective actions should not be implemented solely on the basis of actual or projected doses reaching a predetermined criteria, but rather an integrated response weighting the benefits and shortcomings of protective actions should be carefully evaluated. On the one hand, the justification concept does not advocate for an automatic triggering of protective actions in a given EPZ, but on the other hand, conducting an assessment of and determining a prognosis for an ongoing NPP emergency might delay the response and lead to the late implementation of protective actions.

4.6.13.4. Assessment driven compliance criteria

Assessment of and a prognosis for radiological consequences constitute the basis of many emergency centres in providing inputs that will eventually inform the decision to take off-site protective actions. Close distances to the source of the release mightsee dose values exceed generic criteria in place for taking precautionary urgent or urgent protective actions.

An assessment driven emergency response framework, whereby protective actions are only deployed after conducting assessment of and determining a prognosis for the ongoing accident, relies on two aspects:

— The assessment is reliable enough: the uncertainty embedded in all computational processes involved in a dose projection is quantified or, as a minimum, dose results are subject to be expressed along with uncertainty bands;
— A set of time dependent conditions are met between the assessment, implementation of protective actions and keeping public exposure below the reference levels in place.

4.6.13.5. Timeline layout and related basic concepts

Figures 9(a) and 9(b) show two generic timelines after general emergency declaration. This timeline includes basic parameters helpful to develop a traceable and graded EPZ methodology:

t_{IE} - Initiating event time
t_{GE} - General emergency declaration
$t_{a,b}$ - Dose assessment calculation start time
$t_{PA,b}$ - Beginning of implementation of all protective actions
$t_{r,b}$ - Beginning of radioactive releases
$t_{a,f}$ - Dose assessment calculation end time
$t_{r,f}$ - End of radioactive releases
$\Delta t(PA_i)$ - Elapsed time needed to implement protective action i
$\Delta T(RL_i)$ - Elapsed time for reaching reference level associated with protective action i. This time period takes into account the travelling time from release to exposure point after the start of the release.

This situation shown in Fig. 9(c) applies for all three protective actions.

As stated in para. 5.7 of IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [56], this justification criterion requires that “The government and the regulatory body or other relevant authority shall ensure that the protection strategy for the management of existing exposure situations… is commensurate with the radiation risks associated with the existing exposure situation”. Given the frequency and hazard distribution of some of the SMR technologies, featuring radiological releases far from reference levels for most of the accident’s spectrum, to implement protective actions below but also beyond what needed should be prevented to a reasonable extent[3]. Therefore, the path leading to a tailored emergency response involves supporting the recommendations on actual accident assessments rather than deploying actions solely based on boundary, worst-case sequences, provided the two-abovementioned criteria are met.

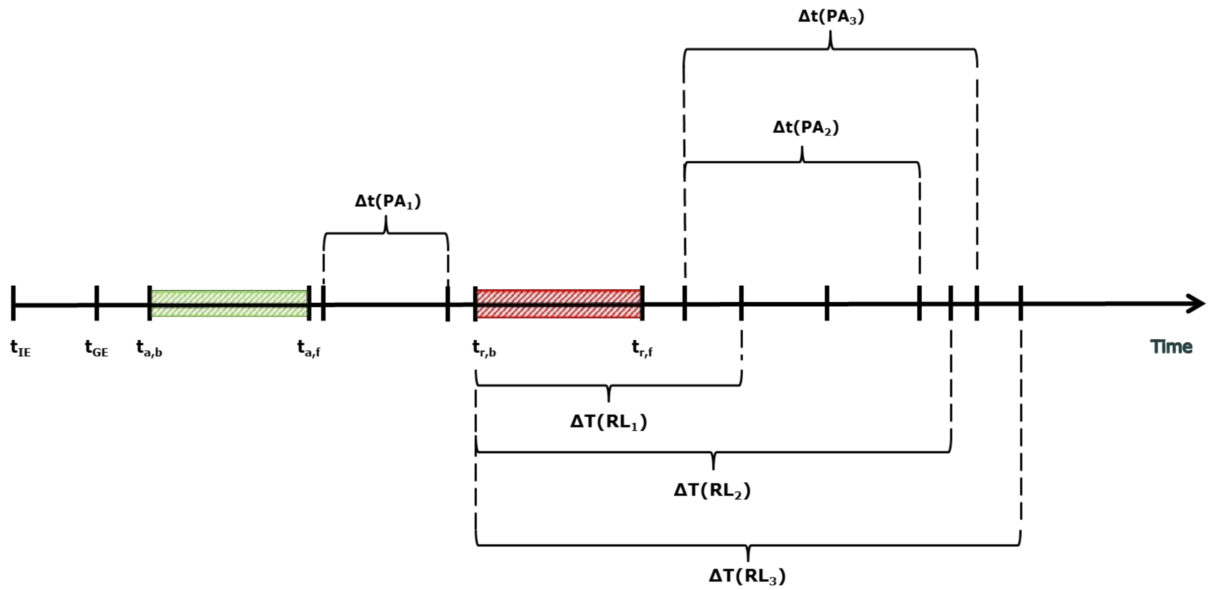

FIG. 9(a). Emergency response generic timeline A.

[3] Such a degree of reasonability relies on the accuracy in the dose consequence predictions, which is at the same time dependent on an accurate estimate of the different uncertainty sources embedded throughout the assessment process.

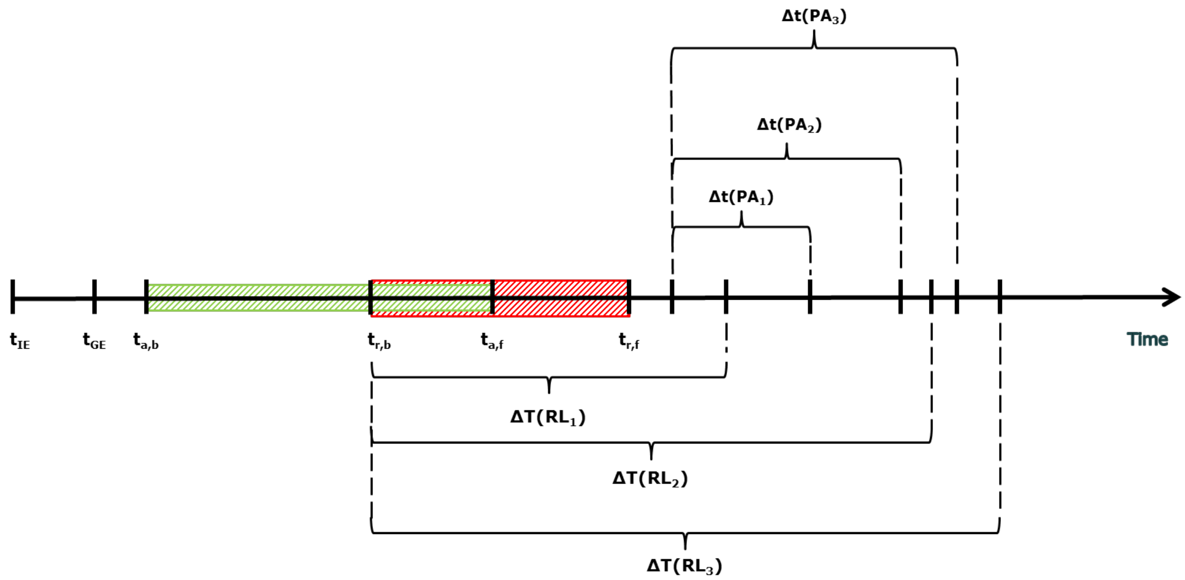

FIG. 9(b). Emergency response generic timeline B.

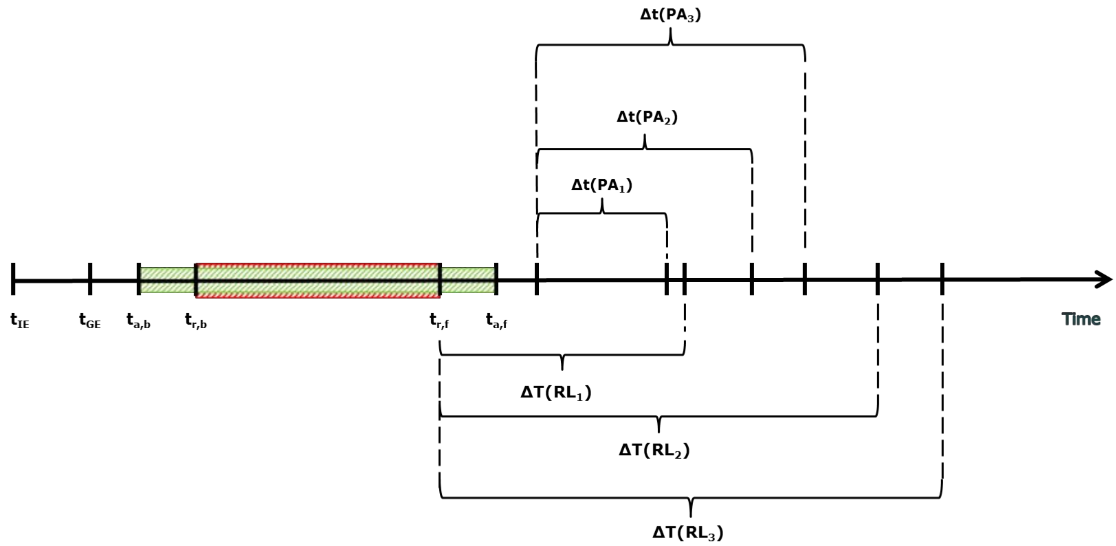

FIG. 9(c). Emergency response generic timeline C.

4.6.13.6. Automatically triggered protective actions in EPZ

Many countries plan to implement protective actions in EPZ without waiting for results from assessment and prognosis of the emergency, including dose projection. These countries highlight:

— The possibility of an early significant release; and/or
— High uncertainties inherent to dose projection. WENRA-HERCA evacuating a specific EPZ immediately after the declaration of a general emergency but only for scenarios featuring an absence of information or reliable assessment [57].

4.6.13.7. Assessment driven EPZ

A more flexible approach consists of defining the EPZ without binding a set of protective actions to a specific precondition.

This emphasizes the need for justification for public protective actions tailored to the actual or anticipated characteristics of a radioactive release. As a result, in recognizing the logistical and difficulties and social burden of implementing protective actions, and the public harm potentially generated in their implementation, no evacuation would be decided without dose projection outputs.

4.6.13.8. Impact of integrating SMR EPZ areas compared to a larger single EPZ

On the one hand, the lower thermal powers featured by SMRs would likely result in smaller EPZ sizes; on the other hand, the deployment of SMRs might result in an overall picture along a territory that differs drastically from the current situation, for example, at country level, where NPPs constitute just a very small number of large production sites. Since multiple nuclear sites to compensate for the same amount of power provided by a single large NPP might be needed, it is unclear how much smaller the EPZ distances for an SMR compared to an existing NPP would be.

Figure 10 shows two NPP deployment situations. On the left, in scenario 1, there is a large NPP whose EPZ radius is taken as 10 km (leading to an EPZ area of 314 km^2), while on the right, in scenario 2, there is a set of SMR sites whose overall featured electric power matches that of scenario 1, with an EPZ radius of 5 km for each SMR — a total EPZ area of 392 km^2. Therefore, even if the EPZ radius of a single SMR is smaller than that for a large NPP, the total EPZ area of the set of SMR sites needed to deliver the same amount of power is larger.

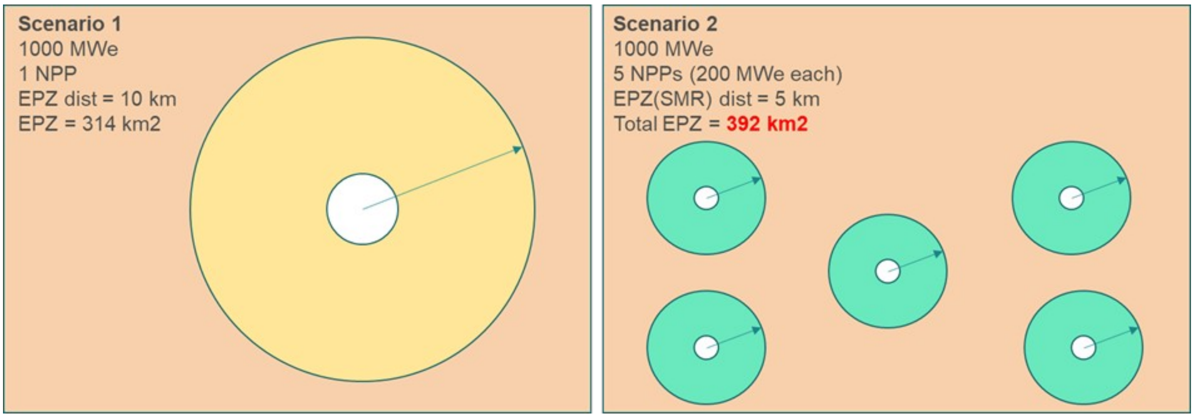

FIG. 10. Comparison of the total EPZ area between a large NPP and five SMR sites featuring the same power output.

This would advocate for scaling based criteria to facilitate comparison of radioactive releases between plants of different sizes (or output powers). Furthermore, given that EPZ sizes are far from being linear with distance from the release point for a particular dose threshold, it is unclear how small the SMR EPZ size compared to the larger NPP EPZ size would be to meet this overall criterion on EPZ areas. A solid elaboration on such scaling criteria is found in [58].

4.6.13.9. Methodology flowchart for the accident selection and source term calculation

The JRC worked on a methodology flowchart for the accident selection and source term calculation based on the JRC findings in this project. This generic methodology grounds on two fundamental aspects driving the accident selection, namely the national regulatory framework and the nuclear reactor design state of the art. The aim of this methodology is to be technology-neutral but also regulatory-neutral to be applicable for all countries and all SMR designs.

4.6.14. Findings from Pakistan Atomic Energy Commission, Pakistan

The work conducted by PAEC allowed to estimate EPZ sizes for a 100 MW(e) iPWR SMR based on scaled down core inventory of a 1100 MW(e) Gen III+ PWR, atmospheric dispersion and dose projection considering various exposure pathways.

The source term initially considered came from [11]. Accordingly, it was assumed that 10% of the volatile fission products (krypton, xenon, iodine, tellurium, caesium) from the iPWR100 core inventory would be released off-site as a single plume at ground level and for a duration of 10 h [11].

The meteorology was based on site specific hourly observations of wind speed, wind direction, stability class and precipitation observed at 10 m height, along with eight mixing heights (four seasons×day/night) for the representative sites were used. The normal calculation mode for MACCS is to sample from hourly weather data.

The first generic criterion expressed as RBE weighted absorbed dose to red marrow ($AD_{\text{red marrow}}$) of 1 Gy was projected to be exceeded at a distance of ~1.1 km for an individual without sheltering in a house, and at ~0.5 km for an individual sheltering in a house.

The second generic criterion of RBE weighted absorbed dose to the foetus (AD_{foetus}) of 1 Gy was projected to be exceeded at a distance of ~0.3 km for a pregnant woman taking an ITB agent before inhalation with sheltering in a house, and at a distance of ~2.1 km if sheltering in a house without taking an ITB agent; further, the same generic criterion was projected to be exceeded at a distance of ~0.7 km when only an ITB is taken and ~3.3 km without house sheltering and ITB.

The generic criterion of 100 mSv of effective dose from inhalation (E_{inh}) was projected to be exceeded at a distance of ~4.8 km for an individual sheltering in a house and at a distance of ~7 km without sheltering in a house.

The generic criterion of 100 mSv equivalent dose to the foetus from inhalation ($H_{\text{foetus,inh}}$), when an ITB agent is taken before or shortly after inhalation, was projected to be exceeded at a distance of about 5 km for a pregnant woman sheltering in a house. The same generic criterion was projected to be exceeded at a distance of ~33 km without house sheltering and ITB, and at a distance of ~8 km when only ITB is taken and ~23 km when sheltering in a house without taking an ITB agent. From these results, the generic criterion regarding the dose to foetus appeared to be limiting.

It was concluded that the optimized PAZ and UPZ sizes for iPWR (10% core volatile release) are 0.3 km and 5 km, respectively, if sheltering in a house and ITB agent taking are in place. Without sheltering and ITB, the PAZ and UPZ sizes would be 3.3 km and 33 km, respectively.

To assess site to site variation, analysis for two representative sites (Site-1: coastal and Site-2: inland) was performed. The weather conditions in the coastal region are much windier. As a result, the PAZ and UPZ sizes that were estimated for Site-2 were almost twice as high as those for Site-1.

As a sensitivity study, considering variation due to different power levels and release fractions, iPWR-50 (50 MW(e)) with 5% releases for Site-1 was also analysed. Considering the generic criterion expressed as dose to the foetus and assuming that sheltering in house and ITB agent taking are in place, the PAZ and UPZ sizes would be reduced compared to iPWR-100 (10% release and Site-1): the PAZ size would decrease from 1.5 km to 0.1 km, and the UPZ size would decrease from 14 km to 2 km.

The reduced source term will result in reduced EPZ sizes and slow accident kinetic progression will likely result in the effective and timely implementation of protective actions. Based on the analysis, the suggested PAZ and UPZ sizes for iPWR 50 with 5% volatile core release, assuming that sheltering in house and ITB agent taking are place, would be 350 m and 2.5 km, respectively.

For SMRs of 50 MW(e), with <5% core volatile (Xe, Kr, I, Cs, Te) release to the environment, the following are the main points concluded on the basis of the CRP work:

- No PAZ would be needed, that is, no precautionary evacuation based on plant conditions before a release would be needed; however, effective/smart evacuation based on online monitoring would be required in the UPZ.
- In the preferential UPZ and wind dominant direction (cone area), the arrangement of large building sheltering and pre-distribution of ITB should be considered at the preparedness stage.
- Evacuation can be delayed and sheltering (for a justified period) with ITB might be considered until monitoring data results are available to further assess the situation and act accordingly. For this a robust monitoring system should be established to ensure the availability of online monitoring data even in severe conditions. The continuation of the initial sheltering in place or the implementation of further protective action should be contingent on the results of the environmental monitoring. Based on monitoring results, evacuation can be carried out on a need basis and effective manner.

4.6.15. Findings from Tunisian Company of Electricity and Gas, Tunisia

Based on the deterministic study conducted for a selected site in Tunisia, STEG concluded that, from a technical perspective, the implementation of EPZ remains necessary, particularly as this would be the first nuclear power plant in the country. To enhance public confidence, STEG proposes the establishment of a 1 km EPZ around the site..

4.6.16. Findings from Corporate Risk Associates Limited, United Kingdom

The first of the research projects carried out by CRA in collaboration with Imperial College London studied the impact in terms of societal consequences of a pure scale plant reduction, assuming that the technology of both plants, a large NPP and an SMR, is the same.

The goal was to compare a nuclear fleet made up of a single large plant against a number of SMR sites featuring the same output energy. Assuming that the power output maintains a linear relationship with the source term, the source term size displays a linear relationship with consequences, and that the accident frequency for an SMR is inversely proportional to the number of reactors deployed, if five SMRs were to be deployed, their individual accident frequency would need to be a fifth of that for an equivalent single reactor. Thus this project aimed to explore whether or not the relation between consequences and source term size does in fact display this linearity.

The parameters of the consequence examined within the project are the deterministic and stochastic health effects as well as the area size and population within a DEPZ in the United Kingdom. The project achieved this via the use of PACE, a new software package that incorporates the advances in the field of nuclear safety and atmospheric dispersion in the industry from its predecessor COSYMA.

The project concluded that while all effects scale down with source term size, there are both linear and nonlinear relationships. Furthermore, the study found that the deterministic effects display a nonlinear trend favouring the reduction of source term size. The deterministic consequential effects are found to fall at a faster rate than the decrease of source term size. By contrast, the stochastic effects displayed a linear reduction with source term size and hence showed no relative advantage or disadvantage compared to smaller source term sizing. Finally, DEPZ measures were found to display a large nonlinear trend that did not favour the reduction of source term size. The variation of source term size led to conflicting safety results. Prioritization of safety factors is beyond the scope of the project. Therefore, an evaluation of safety priorities, as well as quantitative analysis of SMRs, is required to provide a conclusion in this regard. However, the study did find that SMRs have the potential to reduce deterministic consequences as they produce smaller source term sizes as a result of their lower power output.

In conclusion, the main results of this project were:

- There are benefits in terms of deterministic health effects (fatalities, doses);
- There is no change to stochastic health effects (cancer, heritage);
- There is a disbenefit in land and population effects.

The second project compared a large NPP against a MSR SMR design. The resurgence of interest in the molten salt reactor as a novel, advanced reactor concept, characterized by the use of molten salt as the coolant, has led to an interest in the benefits it might hold over more conventional reactors. Interest in its safety advantages led to this project, where it is used to evaluate the harm caused in the event of an accident scenario.

The parameters evaluated for the project were the deterministic and stochastic health effects induced by the source term, as well as the affected area and population within a dose limit. The project used a source term provided by an MSR vendor and compared it with the RC-602 basemat failure release category in the United Kingdom.

The project found that the very design of the MSR makes it inherently safer, but the unpredictability of the application of the linear no threshold model raised questions regarding the use of PACE, or indeed any consequence prediction technique, on a source term that emits such a low dosage.

The main conclusions from this project are as follows:

- A reduced source term arising from the reactor due to the chemical interactions in the coolant; and
- A reduced height and an increased duration of the radiological release due to low pressure in the coolant;
- Leading to an overall benefit in deterministic, stochastic, land and population effects.

A third project aimed to compare the prospect of a nuclear fleet consisting of either only large reactors or only small reactors in the safety case for the United Kingdom's six newly proposed nuclear new build sites. To do so, the Level 3 PSA software PACE was used to model radioactive material distribution in an accident scenario, and to calculate the hypothetical harm caused.

The report also considered the optimization of the PACE runs by reducing the source term input data and sensitivity analysis of the results to meteorological condition time frames.

The results conclude that the large-scale reactor would provide the largest accident consequence, with 508 463 individuals evacuated in the worst case, with a probability of occurrence of once in years. This was compared to the worst case SMR accident, whereby 5454 individuals would be evacuated at a higher probability of occurrence of once in years.

It was calculated that the source term could be reduced to only those radioisotopes that contributed <1% of the predicted dose, whilst providing result convergence and a run time reduced by 20.43%.

4.6.17. Findings from Texas A&M University, United States of America

Research at TAMU explored several phenomena in HTGRs, including:

- The gas mixing inside HTGR buildings;
- The concentration and timescale of the gas mixtures that could be released into the atmosphere through the reactor building when accidents occur;
- The phenomena governing the transport of fission products.

Experiments were performed at TAMU's Reactor Building Facility. The results highlighted the important phenomena inside the reactor building, and produced validation data for computer codes that will make source term predictions. Source term predictions are fundamental for determining the EPZ for modular HTGRs. The research, however, did not address EPZ (either PAZ or UPZ) size directly.

4.6.18. Findings from Argonne National Laboratory, United States of America

ANL's primary research concerned source terms for SFRs. Mechanistic source term (MST) for advanced reactors, including SFRs, was identified by the United States DOE as an issue with high regulatory importance and possible long lead time to closure. To that end, a mechanistic source term trial analysis was performed for a metal alloy fueled, pool type SFR. The results of the MST trial analysis are documented in the openly available report ANL-ART-49 [59].

Off-site consequence evaluation was presented for one scenario in the trial analysis: a large, unprotected transient overpower with degraded negative reactivity feedback. Predictions with the WinMACCS code showed a TEDE value below 25 rem (250 mSv) at distances in very close proximity to the reactor building. The 25 rem value is the basis of several NRC requirements, such as the limit of the exclusion area boundary (25 rem within 2 h from the start of release) and low population zone (during the entire passage of the plume or 30 days). The peak TEDEs were, in fact, below 1 rem (10 mSv) at distances smaller than 100 m.

The MST results will be fundamental to the determination of EPZ. Although successful, the trial analysis illuminated gaps in models and data regarding some phenomena that might result in uncertainties or the use of conservative assumptions that could make the justification of reduced EPZ size difficult. The research did not directly estimate the size of an EPZ for a particular reactor or site.

4.6.19. Findings from Pittsburgh Technical, United States of America

Pittsburgh Technical's research focused on developing the technical basis that is required to support a risk informed approach to sizing EPZs for SMRs correctly. This included new studies of relevant phenomena as well as development of improved methods, models and tools. Specific research outcomes can be separated into the categories of outside containment and inside containment.

For outside containment, efforts were made to enhance atmospheric dispersion predictions in the AERMOD computer code by improving handling of near field effects such as building wake and stack tip downwash. A new consequence estimation tool was also developed to improve evacuation time estimates.

For inside containment, experiments were performed to measure decontamination factors in the relatively large surface area to volume ratio of SMR containments. Based on these measurements, a representative decontamination factor for an iPWR was estimated to be 19.7. This is significantly higher than the value of 1.2 commonly attributed to large LWRs.

The improvements to the atmospheric dispersion models and the reduction to the accident source term that are evidenced by this research are intended to support the technical basis for correctly sizing (i.e. making smaller) EPZs for SMRs. This research did not directly estimate the size of an EPZ for a particular reactor or site.

4.7. DISCUSSION ON THE RANGES OF EPZ SIZE VALUES

Figure 11 shows the EPZ sizes obtained by the participants in logarithmic scale. As some participants did not distinguish between PAZ and UPZ but only estimated an overall EPZ side, values are plotted in perspective to the IAEA technical guidance for UPZ.

It should be noted that the EPZ sizes estimated by CRP participants, as reported in this TECDOC and plotted in Fig. 10, are tentative values obtained in the framework of the CRP and do not necessarily reflect official positions by regulatory bodies in the CRP participants' countries.

The wide spectrum of calculated EPZ sizes — with a maximum UPZ upper to lower bound ratio of 150 for a given SMR thermal power — is much larger than for existing large NPPs; inGS-G-2.1 [8] the IAEA suggests an UPZ size of 5–30 km for a nuclear reactor with a nominal thermal power higher than or equal to 100 MW, hence only a factor of 6.

Such a large scattering in the results can be explained as a consequence of the following main drivers:

— 'SMR' does not refer to one specific design but to a family of designs covering a very broad spectrum of nominal thermal powers, advanced design and modularity. This is not the case for existing large NPPs, featuring quite similar aspects related to the magnitude of the nominal thermal power, design, safety systems and hence potential radiological releases off-site.
— The determination of EPZ sizes for existing large NPPs is mainly driven by generic assumptions on the type of bounding accident or direct assumption of the type and amount of radiological releases. Such a generic approach helps results 'converge' among various counties' national approaches, compared to preliminary EPZ size results obtained by CRP participants.
— For different SMR designs with close nominal thermal powers, discrepancies in features such as fuel composition, coolant type (e.g. water, molten salt, sodium) and technical assumptions on the containment might lead to a wide range of source terms.
— The concept of EPZs varies among Member States, as well as the criteria associated with their size.

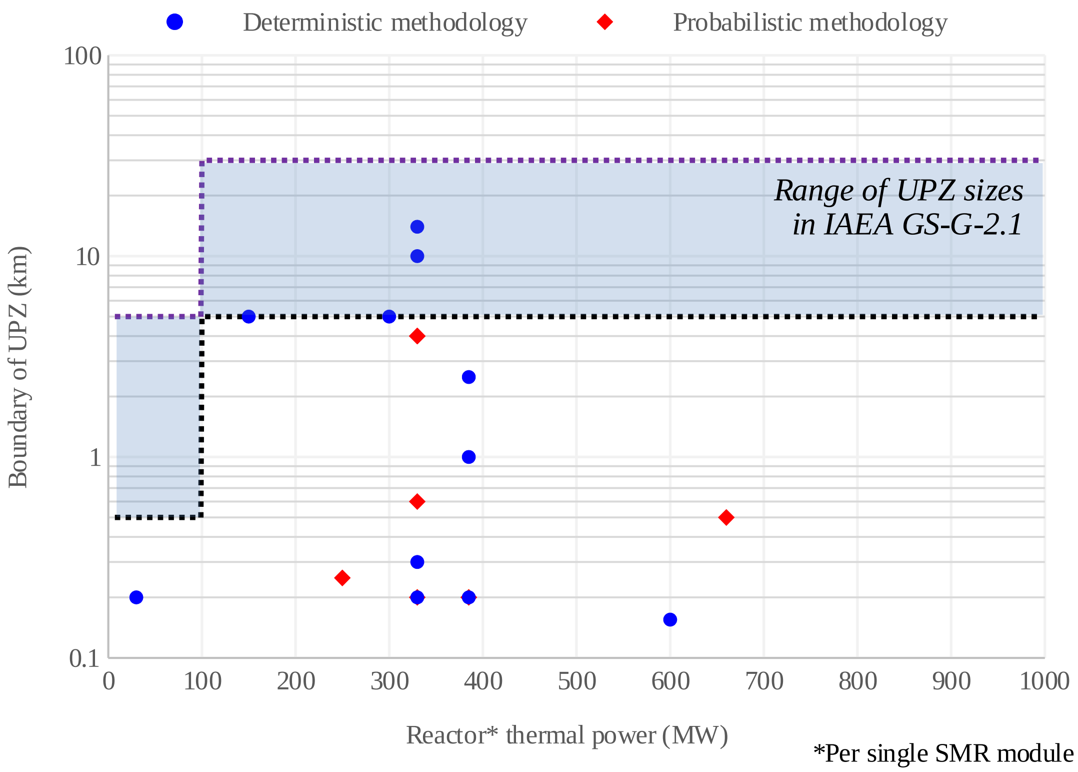

FIG. 11. Overview of the EPZ sizes obtained by CRP participants.

5. SUMMARY AND CONCLUSION

Institutes that participated in the CRP were differently impacted in their research by the coronavirus disease 2019 (COVID-19) pandemic. As a result of the pandemic, a fourth year was added to the CRP, with the fourth and final research coordination meetings (RCM) being held in July 2021. Research contributions to the CRP can be broadly categorized as either addressing perceived deficiencies in the technical basis for SMRs' EPZs or demonstrating a method for determining the EPZ for SMRs via case studies.

In the case of the former, participants examined topics such as risk assessment methodology, source term prediction, and dispersion and dose projection modelling. Several institutes contributed fundamental experimental data and analysis relating to the transport of radionuclides in specific SMR technologies. Novel methods for modelling near field atmospheric dispersion with CFD and three-dimensional dose integration were also developed. Some institutes performed case studies for source term and atmospheric dispersion and dose assessments. These were presented as being important for the determination of EPZs, but definitive conclusions on the necessity or size of an EPZ were not always provided.

Most participating institutes, however, performed case studies that demonstrated the determination of EPZ for a specific SMR technology and site. There is no evidence in the CRP contributions that these institutes conducted a thorough hazard assessment and first assessed the EPC under which the SMR design they considered would be categorized. Instead, each institute considered one or a few specific accident scenarios, which might be seen as boundary scenarios for some categories of hazards. Regardless of the designs and associated nominal thermal powers considered, these institutes initially assumed that an EPZ would be required and worked on sizing it. For that purpose, institutes first selected one or more accident scenarios and related source terms. These accident scenarios included design basis accidents and severe accidents, either built following a deterministic approach or extracted from PSA studies. Then institutes conducted atmospheric dispersion simulations, using Gaussian or Lagrangian models, considering either realistic, average or conservative numerical weather data over various time intervals. Outputs from atmospheric dispersion simulations were used for dose projection, institutes considering multiple exposure pathways, selected population groups and dose conversion factors. Dose projection results were put in perspective with national dose criteria and/or IAEA generic criteria for urgent protective actions to conclude on tentative EPZ sizes.

It needs to be highlighted that many institutes used a probabilistic approach at least for the atmospheric dispersion part of their research. This means that they built a probability distribution of a dose projection to reach a certain national dose criterion, based on several years of observed weather data/conditions. For a given release, this allows one to embrace day/night and seasonal variations of the atmospheric dispersion. This constitutes a change in the overall methodology for sizing EPZs, for instance compared to Appendix I of EPR-NPP-PPA [11], which considers one single atmospheric dispersion scenario in the form of Pasquill stability classes. The progress that has been made on computation capacities over the last decade facilitated performing large probabilistic studies.

Regardless of the methodology used, all EPZ sizes estimated by institutes in the CRP are below the 15–30 km size currently suggested in the IAEA guidance for establishing a UPZ for an LWR with a nominal thermal power higher than or equal to 100 MW(th) [11]. Most EPZ sizes estimated as part of the CRP are smaller than 15 km. The smallest estimates were a few hundred metres around the reactor, which was within what the participants reported as the site boundary. Participants concluded that an EPZ was, therefore, not necessary in such cases. However, these conclusions were not obviously based on a complete hazard assessment and prior determination of an EPC. As per IAEA definitions, EPZs (including PAZs and UPZs) are inherently defined outside the site boundary, and determination of an EPZ distance presumes EPC I or EPC II.

Conclusions on tentative EPZ sizes formulated in the CRP only reflect institutes' views and are mostly based on projected radiological consequences.

6. CONSIDERATIONS FOR FUTURE ACTIVITIES

6.1. SMALL MODULAR REACTORS FOR INDUSTRIAL PURPOSES, NON-ELECTRIC APPLICATIONS — THEIR IMPACT ON HAZARD ASSESSMENT

Due to their lower power output, SMRs are also expected to provide a viable option for partial or full use for non-electric applications closer to users, for example, providing cogeneration of heat for housing or for industrial processes, seawater desalination and hydrogen production. Process heat supply from SMRs through cogeneration is meant to result in significantly improved thermal efficiencies, leading to a better return on investment. Microreactors, a subcategory of SMRs, with power <20 MW(e), might also serve niche markets to replace diesel generators in remote areas or small island locations. Industrial (e.g. the presence of large volumes of hydrogen nearby) and external (e.g. earthquake, flooding, sandstorms, heavy rain) hazards need to be considered when developing EPR arrangements for SMRs used for non-electric applications. When coupling an SMR power plant to a hydrogen plant, for example, the reciprocal effect in the case of an accident in either plant might affect the other one. Potential hazardous events in connection with an industrial process heat application system are being extensively investigated. Such events could include fire and explosion of flammable mixtures, tritium transportation from the reactor core to the product (e.g. hydrogen, methanol), thermodynamic interaction between the SMR plant and the chemical plant, and failures or isolation of the desalination plant, and hydrogen release and toxic gas release. Such external and industrial hazards might require further investigation to inform a graded approach for EPR for those types of SMRs.

6.2. SMR LOCATIONS

6.2.1. Urban environment versus remote areas

Probably the most typical reason to site an SMR plant in a city/urban environment is the production of heat, either for residential heating or for industrial purposes, as heat should not be transported over large distances. Such an environment will have new implications for emergency planning: there are new kinds of hazards and the built environment affects both atmospheric dispersion and external exposure in the event of a release, while the practical feasibility of implementing protective actions is different from that in sparsely populated countryside. Security threats might be more pronounced if the city is targeted for malevolent acts. Industry very nearby might cause new hazards for the SMR, and an accident at the SMR might increase the probability of an industrial accident. The combination of nuclear, industrial and conventional hazards needs to be considered. Regarding the dispersion of radionuclides in the air in case of a release, buildings increase terrain roughness, which might decrease wind speeds locally, but also induce more turbulence (e.g. building wake effects). Furthermore, spread directions might favour street ‘canyons’ and other building wake effects. In modelling of external projected doses, three dimensional cloud shine becomes more important, as well as possible deposition on building surfaces. The practical effort needed to protect the dense population, including buildings for vulnerable groups (e.g. schools, hospitals, elderly homes), might be increased. A high density of population might also have negative impacts on the implementation of public protective actions (e.g. traffic jams). In contrast, remote locations usually have less dense residential areas, but fewer road connections and available shelters that might complicate the implementation of public protection actions.

6.2.2. Underground

Typical reasons to consider underground siting of SMRs include reducing security threats, lack of space to build in a city, or possibly public acceptance. Some historical examples of underground siting are Halden in Norway, Humboldt Bay in the United States, Ågesta in Sweden, Chooz A in France and Lucens in Switzerland. As an advantage, PIEs might have minimized consequences (e.g. aircraft crash, military strike), but more pronounced hazards might arise from fire, flood and availability of the ultimate heat sink. A typical underground space would have partitioning into compartments (possibly including the control room) and an access tunnel that it is possible to close with a certain degree of pressure tightness. Regarding potential pathways for a release to the environment, other connections to the ground surface might be ventilation, electric or a connection to a district heating network. Level 2 PSA (containment events) would possibly include the rock cavern, adding to the slow accident progression kinetics, dilution and decontamination of a release, but release height could be lower than from a high chimney. There might be interrogation regarding the operating organization’s role for emergency response on the surface.

6.2.3. Floating power units (stationary)

An already deployed kind of transportable reactors are floating SMRs. These reactors might be new designs or based on existing marine propulsion reactors. For example, KLT-40S for the Russian floating power unit Akademik Lomonosov (connected to the grid in 2020) was developed based on the experience of operation of KLT-40 reactors on nuclear powered icebreakers Taymir and Vaygatch. A new generation of RITM series reactors are now under operation on 4 icebreakers and are expected to be used in the advanced designs of floating nuclear power units. At present, several projects exist in other countries with new floating reactor designs. Recent development has increased the need to identify, determine and describe the hazards for floating SMRs and related off-site consequences (including for nearby ships or coasts) in case of an emergency. Like other SMR plants, a floating one contains the reactor core radioactive nuclides and possibly spent fuel storage, which might be dispersed in the environment due to an accident (particularly ship collisions), or possibly intentionally, as a result of a malevolent act. As opposed to land based fixed NPPs, a floating SMR could theoretically be towed away from the imminent hazard or threat. Releases might be atmospheric and/or marine (aquatic). In addition, further research might investigate the possibility of relocating a floating SMR unit to decrease off-site radiological consequences in the case of an emergency.

6.2.4. Transportable SMRs (during transport)

Small transportable NPPs are intended to be factory produced, be transportable by truck, rail, aeroplane or ship, and have passive self-controlling features. They might be used for production of electricity and heat in industry and municipal engineering, but also possibly for activities in a remote or isolated area.

SMRs also could be an inherent part of the vehicle, like nuclear propelled ships and floating nuclear power units, where a nuclear reactor is part of the ship's equipment. For these kind of projects nuclear safety needs to be assessed within the safety of the ship as a whole.

For microreactors (typically, nominal power lower than 20 MW(th) or 10 MW(e)), the main plant components could be vehicle mounted. Depending on operating location and transport route, some radiation shielding, adding to weight and size, might be needed around the reactor to protect the public. Transport accidents (e.g. road or rail accidents, air crashes) will need to be considered in the hazard assessment. Regarding emergency planning, there might be situations in which transport through densely populated area cannot be avoided. In addition, research should further investigate the release phenomenology associated to microreactors (i.e. low source term but an almost immediate release). For microreactors and floating SMRs, research might also investigate ways to develop EPR arrangements along the transport route, including in areas/jurisdictions that are not acquainted to the nuclear risk.

6.3. MULTIPLE SMR MODULES COLLOCATED AT A GIVEN SITE

The research contributions from participating institutes almost all (with one exception) considered deployment of a single SMR module at a given site. The impact on the overall EPZ size of having two or more SMR modules collocated on a given site was not investigated. Nevertheless, it is expected that individual sites might contain many SMRs with varying levels of independence, ranging from limited sharing of site services to the interconnectedness of some safety related features. The appropriateness of scaling hazards shared by multiple modules on a single site, and the new hazards introduced by the collocation of multiple modules on a single site, were not investigated by the participants in any significant detail. Further research might investigate the specific hazards introduced by collocating multiple modules (possibly of different technologies) at a single site.

6.4. EMERGENCIES TRIGGERED BY SECURITY EVENTS

As per para. 4.22 of GSR Part 7 [4], "The government shall ensure that the hazard assessment includes consideration of the results of threat assessments made for nuclear security purposes". For all facilities and activities, the hazard assessment needs to be informed by relevant inputs from the threat assessment, so EPR arrangements are developed based on considerations that include nuclear or radiological emergencies triggered by security events. However, as part of the CRP, nuclear emergencies at SMRs triggered by security events were barely considered (only one institute did consider accident scenarios whose triggering events were security related). For future work on SMRs, thorough hazard assessments need to be conducted, with due considerations for emergencies triggered by security events. The whole content of the threat assessment, which might be confidential, does not need to be shared to conduct such a work, only relevant conclusions and inputs from the threat assessment are needed. Security events that might trigger a nuclear or radiological emergency at an SMR might be specific to a geographical zone, an industrial purpose or a given period.

6.5. PROBABILISTIC METHODOLOGY

For sizing an EPZ, a probabilistic approach has the advantage of considering a wide spectrum of accident sequences, from intermediate likelihood and moderate off-site radiological consequences to low or very low likelihood and high off-site radiological consequences.

However, building numerous accident sequences faces inherent numerical uncertainties and challenges in estimating failure probabilities.

For Level 1 PSA in particular, currently operating large LWR plants have a great deal of operating experience and database resources for the failure of systems, structures and components to estimate the failure frequencies leading to fuel damage. However, for many SMR designs, there is experimental testing experience on systems but limited plant operating experience, including for new innovative design features (such as passive safety systems and their failure frequencies).

Further research might:

— Investigate theoretical failure frequencies (using appropriate modelling tools);
— Gather and process future feedback data on failure frequencies from operating SMRs as they become available.

6.6. MODELLING TOOLS

The whole chain of expertise for safety studies, including studies in the area of EPR, relies on the use of multiple types of calculation codes for simulating various accident sequences and phenomena. Several SMR designs that intend to use novel materials, geometries and safety systems raise new questions about the accident phenomenology. Calculation codes might need further research to develop and validate models to simulate specific physical or chemical phenomena in an emergency.

6.7. CHARACTERIZATION AND QUANTIFICATION OF UNCERTAINTIES FOR DECISION MAKERS ON EPR ARRANGEMENTS, INCLUDING EPZs

The scope of emergency arrangements for SMRs might be a step change compared to current practice for large NPPs. Decision makers need to understand the basis for any differences and how these are influenced by plant design and analysis approaches. It would be beneficial to identify and quantify key areas of uncertainty in the analyses that support the development of emergency arrangements; focusing on any factors that might be particularly significant in the context of SMRs. This might identify where outputs are sensitive to key assumptions, leading to the potential for ‘cliff edges’ for key accident events (e.g. the start of fuel degradation). The consideration needs to include factors that affect the type and the size of the EPZs and the extent and timing of protective actions. Examples include:

— The extent to which hazards associated with new/novel plant are understood and have been fully analysed;
— The effect of limited operational experience with SMRs;
— The limitations of models (and their validation) when applied to SMRs;
— The use of margins and conservatisms (to cover the above) and the justification for these;
— Particular aspects of the design that mean that traditional emergency planning assumptions/approaches might not be relevant, or where new approaches might be needed.

The objective would be to provide further information on aspects that might need additional scrutiny by decision makers when looking to review and implement emergency arrangements for SMRs.

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LIST OF ABBREVIATIONS

ADDAM	Atmospheric Dispersion and Dose Analysis Method
ANL	Argonne National Laboratory
ATLAS	Atomic Technologies Licensed for Applications at Sea
BAPETEN	Nuclear Energy Regulatory Agency of Indonesia
BATAN	Nuclear National Energy Agency of Indonesia
BDBA	beyond design basis accident
BRIN	National Research and Innovation Agency of Indonesia
BWR	boiling water reactor
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CMT	condensate makeup tank
CNEA	National Atomic Energy Commission of Argentina
CNL	Canadian Nuclear Laboratories
CNPE	China Nuclear Power Engineering Corporation
CNSC	Canadian Nuclear Safety Commission
CPU	central processing unit
CRA	Corporate Risk Associates Limited
CRP	coordinated research project
DBA	design basis accident
DCF	dose conversion factor
DEC	design extension conditions
DEPZ	detailed emergency planning zone
DiD	defence in depth
DOE	Department of Energy
EAB	exclusion area boundary
EPA	Environmental Protection Agency
EPD	extended planning distance
EPZ	emergency planning zone
EPC	emergency preparedness category
EPR	emergency preparedness and response
EU	European Union
FFT	fast Fourier transform

FHA	fuel handling accident
GPW	ground penetrating weapon
GTHTR300	Gas Turbine High Temperature Reactor 300
HTAIF	High Temperature Air Ingress Test Facility
HTGR	high temperature gas cooled reactor
HTR-PM	High-Temperature Gas-cooled Reactor Pebble-bed Module
HTTR	high temperature engineering test reactor
ICPD	ingestion and commodities planning distance
ICST	in-containment source term
iPWR	integral pressurized water reactor
INET	Institute of Nuclear and New Energy Technology
IRR	individual radiological risk
ITB	iodine thyroid blocking
JAEA	Japan Atomic Energy Agency
KAERI	Korea Atomic Energy Research Institute
K.A.CARE	King Abdullah City for Atomic and Renewable Energy
LBLOCA	large break loss of coolant accident
LFR	lead cooled fast reactor
LOCA	loss of coolant accident
LPZ	low population zone
LWR	light water reactor
MSLB	main steam line break
MSR	molten salt reactor
MST	mechanistic source term
NFFT	non-uniform fast Fourier transform
NNSA	National Nuclear Safety Administration
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
NSCA	Nuclear Safety and Control Act
NSSC	Nuclear Safety and Security Commission
PAEC	Pakistan Atomic Energy Commission
PAZ	precautionary action zone
PGM	polydisperse Gaussian moment

PHWR	pressurized heavy water reactor
PIE	postulated initiating events
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
PSAR	Preliminary Safety Analysis Report
PWR	pressurized water reactor
RBE	relative biological effectiveness
RCPB	reactor coolant pressure boundary
REGDOC	Regulatory document
REPPIR	Radiation (Emergency Preparedness and Public Information) Regulations
RPV	reactor pressure vessel
SAGNE	Standing Advisory Group on Nuclear Energy
SBO	station blackout
SBLOCA	small break loss of coolant accident
SFR	sodium cooled fast reactor
SGTR	steam generator tube rupture
SMR	small modular reactor
SNERDI	Shanghai Nuclear Engineering Research and Design Institute
SODAR	sonic detection and ranging
SSC	structures, systems and components
SST	siting source term
STC	source term category
STEG	Tunisian Company of Electricity and Gas
TAMU	Texas A&M University
TEDE	total effective dose equivalent
TVO	Teollisuuden Voima Oyj
UPZ	urgent protective action planning zone
VTT	Technical Research Centre of Finland
WG	working group

CONTRIBUTORS TO DRAFTING AND REVIEW

Akhtar, S.	Pakistan Atomic Energy Commission, Pakistan
Anderson, J.	International Atomic Energy Agency
Asano, K.	Toshiba Corporation, Japan
Bell, A.	Corporate Risk Associates Limited, United Kingdom
Caputo, M.	National Atomic Energy Commission, Argentina
Constantin, A.	International Atomic Energy Agency
Dakin, R.	Office for Nuclear Regulation, United Kingdom
de La Rosa Blul, J.C.	European Commission — Joint Research Centre at Petten, Kingdom of the Netherlands
Dobbertin, M.	International Atomic Energy Agency
Fang, S.	Tsinghua University, China
Gimenez, M.	National Atomic Energy Commission, Argentina
Hassan, Y.	Texas A&M University, United States of America
Hummel, D.	Canadian National Laboratories, Canada
Ilvonen, M.	Technical Research Centre, Finland
Khalfoun, M.	Tunisian Company of Electricity and Gas, Tunisia
Kim, S-B.	Korea Atomic Energy Research Institute, Republic of Korea
Lebel, L.	Canadian National Laboratories, Canada
Mohammed Natto, O.	King Abdullah City for Atomic and Renewable Energy, Saudi Arabia
Na, X.	China Nuclear Power Engineering Corporation, China
Ohashi, H.	Japan Atomic Energy Agency, Japan
Pande, M-U.	National Research and Innovation Agency, Indonesia
Pereira Campos, V.	International Atomic Energy Agency
Pistinner, S.	Soreq Nuclear Research Centre, Israel

Reitsma, F.	International Atomic Energy Agency
Salnikova, N.	Afrikantov OKBM, Russian Federation
Setiawan, M.B.	National Research and Innovation Agency, Indonesia
Sidilkover, D.	Soreq Nuclear Research Centre, Israel
Stephani, F.	International Atomic Energy Agency
Subki, H.	International Atomic Energy Agency
Talabi, S.	Pittsburgh Technical, United States of America
Wang, X.	Shanghai Nuclear Engineering Research and Design Institute, China
Zakaria, N.	International Atomic Energy Agency

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