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Emergency planning zones for small modular reactors

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Abstract

The objective of this study is to present an overview of the regulations related to emergency planning and response (EPR) and more specifically of the Emergency Planning Zones (EPZ). Many contemporary developers of small modular reactors (SMRs) aim for designs that do not require off-site EPZ which would reduce the burden on the population living in the vicinity of an SMR allowing, at the same time, SMRs to be located closer to the end-users. To achieve this, SMR vendors need to justify and demonstrate the plant safety adhering to national and international regulations. Details of the SMR developers’ plans to achieve this goal is studied in this report. The study focuses on a set of SMRs under development in the US, Canada and United Kingdom and discusses the regulatory regimes in these countries. A real-world example of an early site permitting process in the US is presented.

A literature review if carried out for current practices and regulations on radiation protection and emergency preparedness and planning (including EPZs) in different regulatory regimes. General overview of the source term estimation, one of the key elements in EPZ determination, is provided. Various considerations for SMRs to implement performance-based and risk-informed regulatory compliance approaches compared to traditional, prescriptive methods are presented and discussed throughout the paper.

1. Introduction

European Union is fighting climate change by setting a target of becoming carbon neutral by 2050 and carbon negative after that to remedy the effects of already emitted greenhouse gases. Energy sector is a major source of one the main greenhouse gas, CO2, emissions (see Figure 1).

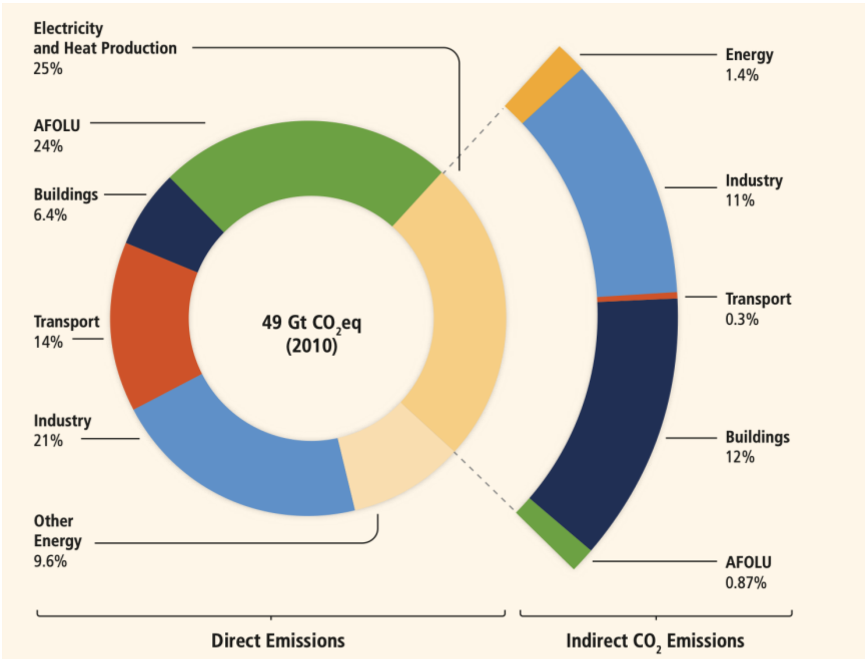

Figure 1: CO2 emissions by sector[1].

In order to transform the energy sector at large scale, it is crucial to replace carbon intensive fossil-based energy production with clean sources. According to the IPCC, the energy sources with lowest carbon emissions are renewables like hydro, solar and wind and nuclear indicating that given the urgency of climate change it is those clean sources that have a role to play.

Nuclear reactors contribute already today greatly to avoiding CO2 emissions worldwide. In Europe, about 100 nuclear reactors provide about 25% of produced electricity and between 40% and 60% of low carbon electricity every day (Figure 2).

[1] IPCC, "AR5 Climate Change 2014", https://www.ipcc.ch/report/ar5/wg3/

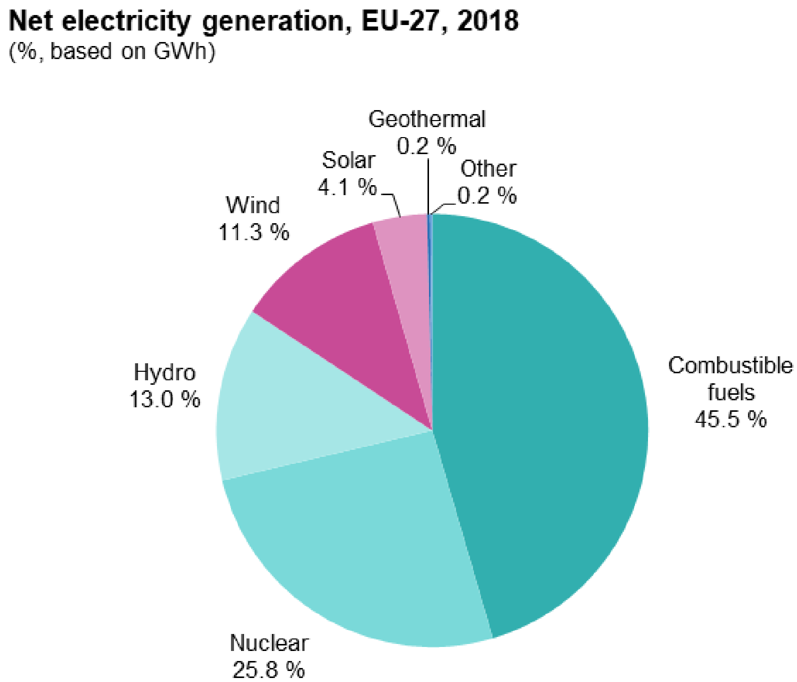

Figure 2: Electricity production mix in Europe according to Eurostat.[2]

Today, reactors derived from designs originally developed for the naval fleet generate 85% of the world's nuclear electricity. Compared to the marine counterparts, the commercial reactors are really big in size. This means longer build times, bigger investments and often going over budget.

In the last decade, there has been an increasing interest in small modular reactors (SMRs). Characterized by their small size, passive safety features and apparent lower financial commitments, SMRs are an attractive alternative to large reactors for countries looking to expand their nuclear fleet and new entrants, such as Estonia.

SMRs provide an option to fulfill the need for flexible power generation for a wide range of users and applications. They are based on advanced nuclear technologies and could be deployed as single or multi-module plants offering the possibility to combine nuclear with alternative energy sources, including renewables.

Typical features of SMRs:

- Electric power up to 300 MWe

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[2] EC Eurostat website, https://ec.europa.eu/eurostat, (Accessed Aug 1, 2020).

- Enhanced safety by eliminating many sources of risk and potential initiating events using passive operational and safety features. This contributes to the extended "grace period" (no operator action) and "coping times" (time before depletion of onsite resources)
- Designed for commercial applications including electricity production, desalination, district heating, hydrogen production, process heat production
- Possibility to have multiple units in the same infrastructure
- Cooled by water, gas, molten salt or liquid metal
- Utilizing agile and harmonized licensing frameworks of different nuclear regulators
- Smaller or no off-site emergency planning zones

The specific design, safety and siting features as well as applications of SMRs require dedicated assessment to formulate and plan for adequate Emergency Preparedness and Response (EPR) arrangements, in particular the size of Emergency Planning Zones (EPZs). EPR is one of the elements that has to be addressed when developing a national nuclear program. EPZ is an area around a nuclear facility where arrangements have been made to take adequate actions in the event of an emergency to avoid or minimize severe deterministic effects off the site and to avert doses off the site in accordance with international safety standards. EPZ shall not be considered as a design requirement as they are neither defined nor determined in/by the design. The need for and size of the EPZs is effectively determined by:

- Radiation protection regulation (dose limits to public and professional workers in normal operation and during emergencies)
- Reactor and plant design characteristics (amount of radioactive material, pathways and timing of release, safety systems, etc.)
- Site characteristics (topography, meteorology, (hydro)geology, population, etc.)
- Public behavior (protective action strategies for the doses and exposure pathways)

The main objective of this report is to provide the reader: a comprehensive overview of the EPZ-relevant regulations, the assumptions and methodologies used by different SMR vendors to justify and demonstrate that no off site EPZ is needed i.e., EPZ is limited to site boundary.

An overview of the radiation protection regulations applied to decision making whenever risk of ionizing radiation exposure exists is provided in Chapter 2. Chapter 3 of the report provides an overview of the source term which is a measure to characterize the release of radioactive material to the environment during an accident and estimate off-site consequences. Chapter 4 of the report continues with description of regulatory frameworks and EPZ guidelines implemented in different countries. This is an area where the radiation protection regulations and source term methodologies meet in order to demonstrate the safety and determine the size of an EPZ. The current status of the EPZ-related matters for a selected SMR technologies is provided in Chapter 5. Finally, Chapter 6 concludes the study with the discussion and synthesis on the regulations and activities of different SMR vendors.

2. Radiation protection

Motivation and goals

Regulation is necessary, or required in accordance with international agreements, in order to protect people from harmful levels of exposure to radiation (i.e., radiation dose) and guide decision making. This is achieved using highly conservative "reference levels" and limits of exposure, whose safety is agreed upon by experts in the field of radiation protection. The International Commission on Radiological Protection (ICRP) is the most important body representing these experts and the main source of recommendations on limits of exposure. Its recommendations are typically adopted by the IAEA guidelines and national regulatory agencies.

For routine nuclear facility operations, regulations typically limit public doses to 1 mSv per year, with the additional requirement that doses are kept as low as reasonably achievable (ALARA) taking socio-economic factors into account. In exceptional events (i.e., nuclear accidents), the risks of increased exposure should be compared to the risks of alternatives, such as a stressful evacuation, which can be more harmful to public health when expected doses are low.

These are the principles that will steer emergency planning, response and recovery in the unlikely case of a nuclear emergency.

Doses and limits

A reference level is defined by the ICRP as "the level of residual dose or risk above which it is generally judged to be inappropriate to allow exposures to occur". These reference levels are presented as "bands"/ranges of doses for different types of situations, allowing some flexibility for decisions on an appropriate level of exposure taking other non-radiological considerations into account. Reference levels are expressed in millisieverts (mSv – either acute or per year) and in terms of residual dose – the dose received after any protective actions have been implemented.

Total effective dose limits are suggested by guiding bodies and enforced by national regulatory agencies to individual members of the general public as well as to professional radiation workers.

The IAEA and national regulators typically follow the guidelines on dose limits from the ICRP. According to the latest ICRP recommendations[3] and IAEA safety guides[4], the effective dose a general public individual receives in a year has to be below 1 mSv. The dose limit for professionals

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[3] ICRP, "Annals of the ICRP: The 2007 Recommendations of the International Commission on Radiological Protection", ICRP Publication 103, LINK

[4] IAEA, Radiation Protection of the Public and the Environment, IAEA Safety Standards Series No. GSG-8, IAEA, Vienna (2018)

in normal situations shall be 1 to 20 mSv per year. And the reference level for the public in an emergency situation is 20 to 100 mSv. Some regulations, such as in Canada[5] and in EU[6] allow occupational limits up to 50 mSv/year in certain circumstances but on a condition that the annual average for a five-year period cannot exceed 20 mSv/year.

The US NRC mostly follows ICRP's recommendations, agreeing on the limit to general public[7] but diverging on the professional one[8]. The NRC kept an annual dose limit of 50 mSv believing that a reduction was not urgently required since the average annual radiation dose to occupational workers in 1987 was already below 20 mSv, which is ensured by applying the ALARA.

Estonian regulations[9] strictly follow the Council of Europe directive 2013/59/Euratom, based on the ICRP recommendations. Euratom's limits and ICRP's recommendations agree, for instance, that exposure for professional workers in a radioactive environment, such as a nuclear power plant, should be 100 mSv in a cumulative five-year period while not exceeding 50 mSv in any single year. However, each Member state can still decide on the annual dose limits themselves.

These limits apply only in planned exposure situations but not to medical exposures of patients or in emergency exposure situations where an informed, exposed individual is engaged in life-saving actions or is attempting to prevent a catastrophic situation. In that case the normal dose restriction may be relaxed.

The medical community argues that a change in the dose limit will impact directly the delivery of patient care, suggesting that interventional radiologists and cardiologists may exceed a 20 mSv annual dose limit.[10]

According to UNSCEAR (United Nations Scientific Committee on the Effects of Atomic Radiation), the average annual effective dose to an individual is 3 mSv (out of which 2.4 mSv comes from natural background and the rest from artificial sources, mostly in medicine). In Estonia, the annual effective dose is estimated closer to 3 mSv originating from mostly radon but also from the oil shale industry, medical procedures and traces of Chernobyl fallout.

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[5] CNSC, Nuclear Safety and Control Act: Radiation Protection Regulations, SOR/2000-203
[6] Council Directive 2013/59/Euratom
[7] NRC, 10 CFR Part 20 Subpart D, Radiation Dose Limits for Individual Members of the Public
[8] NRC, 10 CFR Part 20 Subpart C, Occupational Dose Limits
[9] EV Riigi Teataja, “Kiirgustöötaja ja elaniku efektiivdoosi ning silmaläätse, naha ja jäsemete ekvivalentdoosi piirmäärad”, (2018)
[10] W. DA. Mairs, "Occupational dose constraints for the lens of the eye for interventional radiologists and interventional cardiologists in the UK, " Br J Radiol. June 2016; 89(1062): 20150551.

3. Source term

Source term is by definition the types and amounts of radioactive or hazardous material released to the environment following an accident. It is defined as the magnitude, composition, form (physical and chemical) and mode of release (puff, intermittent or continuous) of radioactive elements (fission and/or activation products) released during a reactor accident.

The mechanism, time and location of the release must also be identified. The radiological consequences can be grouped into the following categories:

- Consequences inside the reactor building with doses to operating staff or personnel within the building
- On-site consequences (outside the reactor building)
- Off-site consequences (to members of the public) from:
    a. External exposure from expelled material
    b. Internal exposure from inhaled and ingested active materials

The source term evaluation has a significant weight in the licensing of an NPP. In meeting the requirements for reactor safety, one of the initial steps is to determine the postulated initiating events (PIEs). The PIEs define the scope of the accidents to be used in the safety analysis. The range of PIEs must cover all credible accidents that could have an influence on the safety of the reactor.

One approach commonly used in the reactor safety analysis is to assume a hypothetical accident that results in a bounding source term i.e., one leading to the most severe consequences. An alternative approach is to perform a detailed assessment of accident progression for a number of accident scenarios to derive several different source terms.

Several different approaches can be followed for source term derivation, ranging from a purely deterministic approach through a combination of deterministic and probabilistic approaches to a fully risk-based approach.

Deterministic Approach

Deterministic techniques are typically conservative methods that overestimate the consequences of a radioactive release. They provide a reasonable degree of belief that the ultimate objective of determining a bounding source term can be achieved without performing complex probability calculations. Deterministic estimates can therefore be either conservative or best-estimate together with uncertainty quantification.

The most severe releases (arising from either a DBA or a BDBA) are taken into account in site selection or in setting the design requirements for the engineered safety features (ESFs) of the reactor. These releases may also be used for the purpose of emergency preparedness.

In this approach, the choice of accidents to be considered is based on experience and engineering judgement, without taking into account the probabilities associated with the event sequences, which are necessary for defining the concept of risk associated with the operation of a particular reactor.

Probabilistic Approach

The probabilistic approach assumes that all reactor accidents are possible and that any number of simultaneous failures may occur even though their probabilities of occurrence may be very low. PIEs are used to establish event trees for all possible accident sequences. By quantifying the event trees, one can rank these sequences according to their frequency of occurrence and determine their source terms.

This method takes into account that some accidents or combinations of accidents may have less serious consequences than those used in the deterministic approach, however when weighted by their likelihood, may represent an unforeseen risk and impose different demands on the reactor design.

The probabilistic approach uses the techniques of probabilistic safety assessment (PSA), which:

- Identify accident sequences that may be derived from a broad range of PIEs
- Lead to significant improvements in the understanding of system behavior and interactions, and of the role of operators under accident conditions
- Quantify the risk of reactor operation to the environment, to the public and to site personnel

Derivation of the source term

The release of radioactive substances from a reactor to the environment (the source term) depends on, at least, the following factors:

- The inventory of fission products and other radionuclides in the core
- The progression of core damage
- The fraction of radionuclides released from the fuel and the physical and chemical forms of released radioactive materials
- The retention of radionuclides in the primary cooling system
- The performance of means of confinement
- The release mode (single puff, intermittent, continuous) and the release point (stack, ground level, confinement bypass)

Conservative assumptions will greatly simplify the calculation effort, but often lead to predictions with unrealistically severe consequences. In contrast, realistic assumptions will usually result in source terms that properly reflect the consequences but require elaborate data, calculations and effort. This is particularly true for determining fission product releases from core, primary cooling system and reactor building.

The source term calculations commonly focus on the gaseous, volatile and semi-volatile nuclides since these are the most likely to be released from damaged fuel elements. Precursor sources of radionuclides of interest, such as iodine, can be determined from their decay chains and yields. Precursor sources are important under certain circumstances. For example, the post-shutdown production of I-131 from Te-131 and the production of Xe-135 from I-135 and Te-135 are of importance and should be considered.

The dispersion and deposition of material released to the atmosphere are typically modelled as a plume. Simple plume models can simulate phenomena such as buoyant plume rise, wake effects on plume dispersion caused by obstructions, such as buildings, and wet and dry deposition. Time dependent radioactive buildup and decay in the plume can also be calculated.

4. Regulations

At the highest international level, the IAEA provides the recommendations and guidelines for the EPR. There are intermediate level guidelines or regulations stated by, for example, supranational bodies (NEA, EU). Ultimately, it is the responsibility of each country to decide which guidelines to follow and adopt in their national legislation and regulation in order to protect the people and the environment.

This chapter provides an overview of the EPR guidelines internationally and approaches used in the countries where the reactors of interest of this study are being licensed i.e., United States, Canada and UK. Finally, we present the current status in Estonia.

International organizations

International Atomic Energy Agency

The International Atomic Energy Agency (IAEA) is a United Nations organization established in 1957 to promote safe, secure and peaceful use of nuclear technologies. In 2019, IAEA had 171 member states and Estonia joined IAEA on January 31, 1992.

IAEA establishes international standards issued in the **IAEA Safety Standards Series (SSS)**. This series consists of **Safety Fundamentals** which are met through **Safety Requirements**. These are further divided into **General Safety Requirements (GSR)** and more detailed **Specific Safety Requirements (SSR)**. Guidance on how to meet the Safety Requirements are provided in the related **Safety Guides (SG)**.

The safety standards are recommended for use by member states, national authorities and other international organizations. The IAEA promotes adherence to and implementation of international legal instruments on nuclear safety adopted under its auspices through multilateral conventions. Of particular interest for this study is the Convention on Nuclear Safety which Estonia has been a signatory member since 2006. This means Estonia is obliged to implement certain safety rules and standards at all civil facilities related to nuclear energy including issues of site selection; design and construction; operation and safety verification; and emergency preparedness.

The fundamental safety objective of the IAEA is "*to protect people and the environment from harmful effects of ionizing radiation*"[11]. The Principle 9 in the IAEA Safety Fundamentals states the primary goals of emergency preparedness and response (EPR):

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[11] Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1, IAEA, Vienna (2006)

- To ensure that arrangements are in place for an effective response at the scene and, as appropriate, at the local, regional, national and international levels, to a nuclear or radiation emergency
- To ensure that, for reasonably foreseeable incidents, radiation risks would be minor
- For any incidents that do occur, to take practical measures to mitigate any consequences for human life and health and the environment

The licensee, the employer, the regulatory body and appropriate branches of government have to establish, in advance, arrangements for preparedness and response for a nuclear or radiation emergency at the scene, at local, regional and national levels and, where so agreed between States, at the international level.

The scope and extent of arrangements for EPR have to reflect:

- The likelihood and the possible consequences of a nuclear or radiation emergency
- The characteristics of the radiation risks
- The nature and location of the facilities and activities

Such arrangements include:

- Criteria set in advance for use in determining when to take different protective actions
- The capability to take actions to protect and inform personnel at the scene, and if necessary, the public, during an emergency

The nuclear or radiological EPR requirements which take into account the latest experience and developments in the field are presented in the IAEA SSS No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency[12]. This publication incorporates the findings and recommendations of international expert bodies, notably the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the International Commission on Radiological Protection (ICRP). Adherence to and fulfilment of these requirements, applied by the government at the national level, intends to mitigate the consequences of a nuclear or radiological emergency. Fulfilment of these requirements also contributes to the worldwide harmonization of the EPR arrangements.

These requirements apply to any facility, activity or source with the potential to cause radiation exposure, environmental contamination or concern on the part of the public warranting protective actions and other response actions, irrespective of the initiator of the emergency.

[12] Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSR Part 7, IAEA, Vienna (2015).

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

The goals of emergency response are:

- To regain control of the situation and to mitigate the consequences
- To save lives
- To avoid or minimize severe deterministic effects
- To render first aid, to provide critical medical treatment and to manage the treatment of radiation injuries
- To reduce the risk of stochastic effects
- To keep the public informed and to maintain public trust
- To mitigate, to the extent practicable, non-radiological consequences
- To protect, to the extent practicable, property and the environment
- To prepare, to the extent practicable, for the resumption of normal social and economic activity

It is a responsibility of the national government, regulatory body and operating organization to implement and verify compliance to those requirements so that the EPR goals are fulfilled. A *graded approach* is recommended for EPR arrangements based on assessment of potential hazards and consequences.

For example, the hazards are divided into five categories according to the required EPR and nuclear power reactors belong to the highest, Category I. For the facilities in Category I arrangements shall be made for effectively making decisions on the site and taking precautionary urgent actions, urgent protective actions, early protective actions and other response actions off the site.

The government shall develop, justify and optimize a protection strategy in the preparedness phase for the effective deployment of the protective and other response actions. An important consideration of the strategy is to avoid or minimize severe health effects on the basis of radiation dose. It is advisable to use a reference level for effective dose in the range of 20-100 mSv. The justified and optimized protection strategy shall foresee the implementation of appropriate response actions in case national generic dose criteria are exceeded. This includes establishment of operational criteria (conditions on the site, emergency action levels (EALs) and operational intervention levels (OILS)) for initiating the protection plan.

For emergencies in Category I facilities i.e., nuclear power plants, *emergency planning zones and emergency planning distances* shall be defined for effective emergency response. An emergency at a nuclear power plant that involves fuel damage in the reactor core or in a spent fuel pool can cause death, severe health effects and psychological effects. It can also have economic and sociological consequences affecting the public. Radioactive material from damaged fuel released into the atmosphere will form a plume that is distributed according to the local weather conditions. For effective protection, actions need to be implemented timewise promptly (before the beginning of a severe release) and distance wise gradually (first for those located close to the plant, followed by those further away and so on). Appropriate response activities in an emergency at a light water reactor are described in detail in here[13].

On EPZ

The IAEA defines four different regions around a nuclear power plant for which emergency response planning procedures have to be prepared in advance - two emergency planning zones (EPZs) and two emergency planning distances (EPDs). These regions are illustrated in Figure 3 and described in Table 1.

The two inner regions or EPZs, are the precautionary action zone (PAZ) and the urgent protective actions planning zone (UPZ). PAZ is the area where arrangements have to be made for actions that need to be initiated immediately after the declaration of General Emergency and before the start of a release. UPZ is an area with similar requirements as in PAZ but the actions have to be initiated before or shortly after the release in such a way as not to delay the implementation of the protective response actions within the PAZ.

The two outer regions or EPDs are the Extended Planning Distance (EPD[14]) and Ingestion and Commodities Planning Distance (ICPD). In the EPD arrangements are made to minimize inadvertent ingestion and carry out monitoring to locate hotspots. In the ICPD, arrangements have to be made to protect, monitor and analyze food and water supply.

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[13] IAEA, "Actions to Protect the Public in an Emergency due to Severe Conditions at a Light Water Reactor", EPR-NPP-PPA, Vienna (2013)

[14] EPD is an abbreviation of both, Emergency Planning Distance and Extended Planning Zone. Have to be used with care.

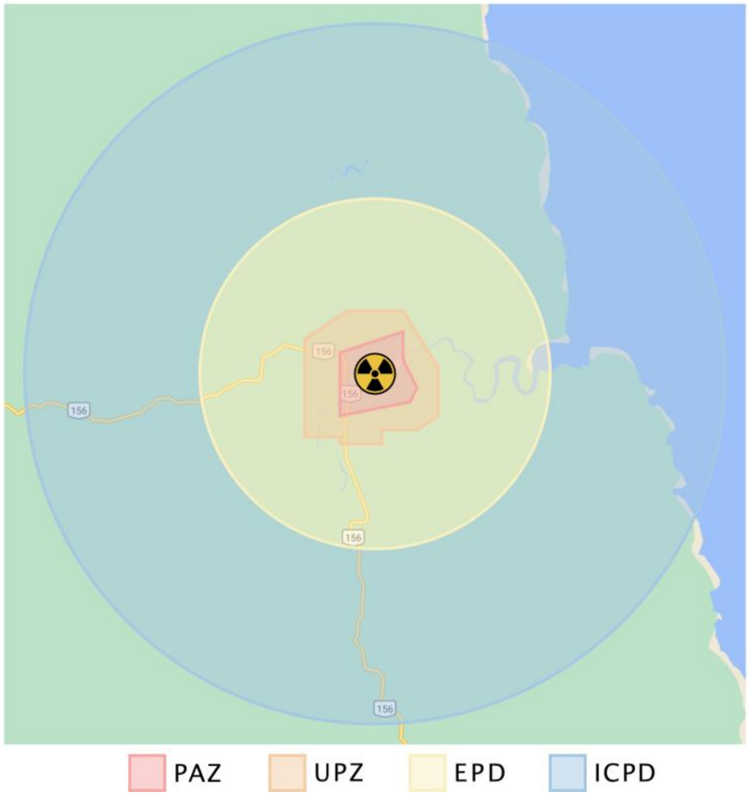

Figure 3: Emergency zones and distances recommended by IAEA.

Table 1: Description of emergency zones and distances.

Emergency Zone/Distance	**Description**
Precautionary Action Zone (PAZ)	An area where comprehensive arrangements are made at the preparedness stage to notify the public and have the public start to take urgent protective actions and other response actions within one hour of the declaration of a General Emergency by the shift supervisor of the nuclear power plant. The goal is to initiate protective actions and other response actions *before the start of a release* warranting protective actions off the site, in order to prevent severe deterministic effects. The boundary of the PAZ needs to be established to minimize evacuation times and evacuation of the PAZ to beyond the UPZ is given priority over evacuation of the UPZ. In addition, provisions are made within this zone for the protection of personnel staffing special facilities such as hospitals, nursing homes and prisons that cannot be immediately evacuated.
Urgent Protective action planning Zone (UPZ)	An area where comprehensive arrangements are made at the preparedness stage to notify the public and have the public start to take the urgent protective actions and other response actions within about one hour of the declaration of a General Emergency by the shift supervisor. The goal is to initiate protective actions and

	other response actions *before or shortly after the start of a release* warranting protective actions off the site, but in such a way as not to delay the implementation of the urgent protective actions and other response actions within the PAZ. In addition, provisions are made within this zone for the protection of personnel staffing special facilities such as hospitals, nursing homes and prisons that cannot be immediately evacuated.
Extended Planning Distance (EPD)	The distance to which arrangements are made at the preparedness stage so that upon declaration of a General Emergency: (a) instructions will be provided to reduce inadvertent ingestion; and (b) dose rate monitoring of deposition conducted to locate hotspots following a release which could require evacuation within a day and relocation within a week to a month. Evacuation of patients and those requiring specialized care would be to locations outside of the EPD to ensure that further evacuations would not be required after a release.
Ingestion and Commodities Planning Distance (ICPD)	The distance to which arrangements are made at the preparedness stage so that upon declaration of a General Emergency instructions will be provided to: (a) place grazing animals on protected (e.g. covered) feed, (b) protect drinking water supplies that directly use rainwater (e.g. to disconnect rainwater collection pipes), (c) restrict consumption of non-essential local produce, wild-grown products (e.g. mushrooms and game), milk from grazing animals, rainwater and animal feed, and (d) stop distribution of commodities until further assessments are performed. The ingestion and commodities planning distance is also the distance within which arrangements are made at the preparedness stage to collect and analyze, during the emergency, samples of local produce, wild-grown products (e.g., mushrooms and game), milk from grazing animals, rainwater, animal feed and commodities to confirm the adequacy of controls.

A methodology that IAEA uses to determine the EPZ sizes can be found in the Appendix I, "Basis for the Suggested Size and Protective Actions within the Emergency Zones and Distances", of this document[15]. The methodology is illustrated in a flowchart in Figure 4. (The EPDs are not determined with this method as the EPDs are rather flexible because of the nature of SMR designs and national regulatory frameworks.)

The EPZ methodology splits into two parts – site and plant. Site part also incorporates the fuel storage and transportation routes. The process takes into consideration the site meteorology using recent data from the nearest weather stations.

[15] IAEA, "Actions to Protect the Public in an Emergency due to Severe Conditions at a Light Water Reactor", EPR-NPP-PPA, Vienna (2013)

Plant design considers the characteristics of the plant such as reactor technology, number of reactors. It can also consider the major design features typical to SMRs including underground placement, modularity, novel features as well as the reduced source terms. Based on this information safety analysis is performed accounting for all potential emergency situations involving severe damage to reactor fuel resulting in radioactive releases and calculating respective doses. Thereafter a heatmap of maximum and average doses will be created based on which a rough estimate of the EPZs can be laid down. Considering the dose calculations, local infrastructure characteristics and other factors that could affect plant safety, the EPZ size is determined and addressed to the public.

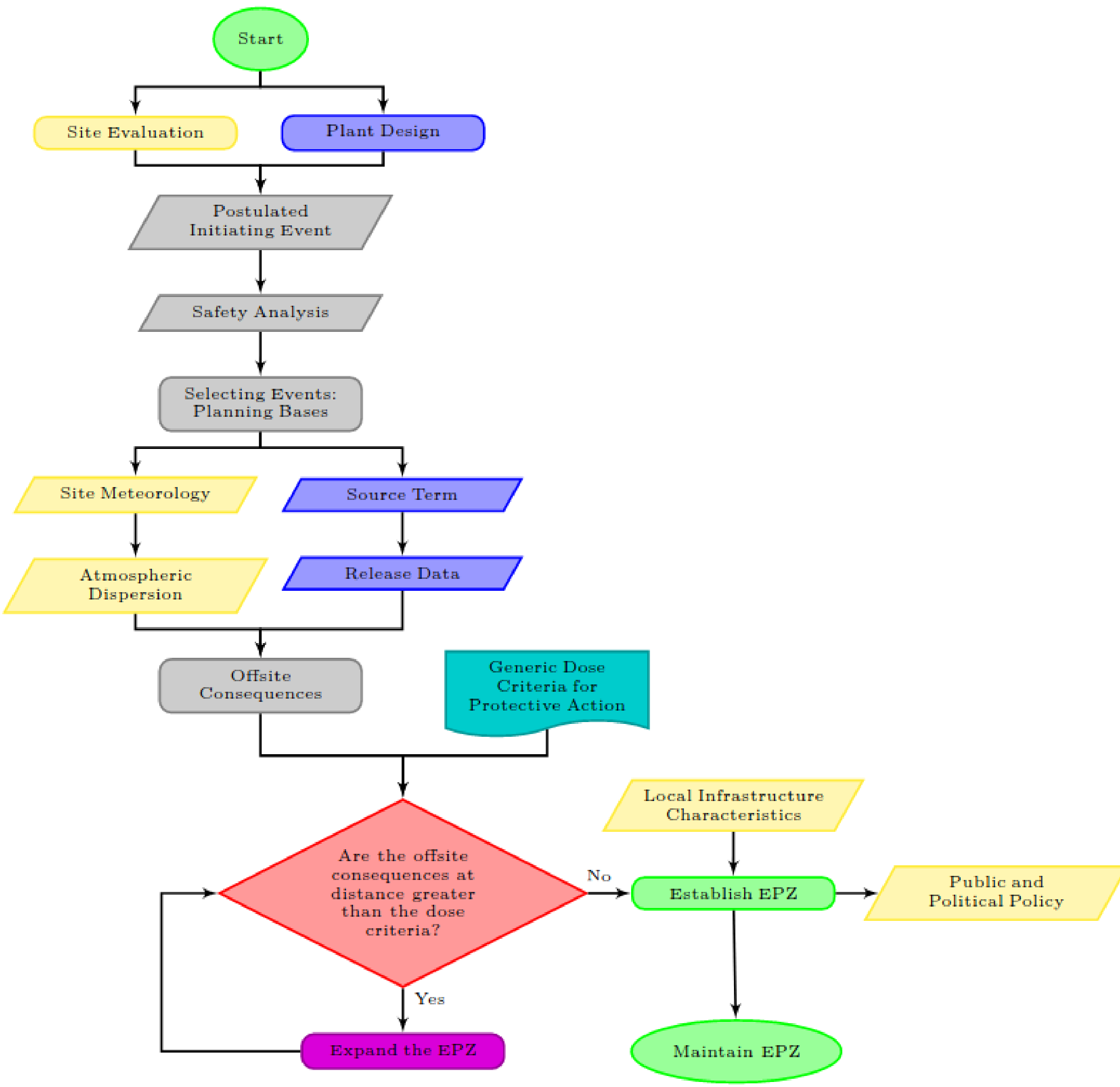

Figure 4: Generalized approach to determine EPZ sizes.

The suggested sizes for EPZs and EPDs are shown in Table 2.

Table 2: Suggested sizes for emergency zones and distances.[16]

Emergency zones and distances	Suggested maximum radius (km)	
	100 to 1000 MWth	≥ 1000 MWth
Precautionary Action Zone (PAZ)	3 to 5	
Urgent Protective action planning Zone (UPZ)	15 to 30	
Extended Planning Distance (EPD)	50	100
Ingestion and Commodities Planning Distance (ICPD)	100	300

In practice, different IAEA member states use somewhat different definitions of emergency planning zones and distances as well as apply different methodologies to estimate their sizes.

Considerations for SMRs

Regulators are being approached by designers with SMR safety case proposals that are seeking to relax regulatory requirements for design and safety analysis. This calls for adequate regulatory regimes including risk-informed, performance-based graded approaches.

In March 2015, the IAEA established an SMR Regulators' Forum. The Forum carried out a study in 2015-2017 to identify and address the regulatory challenges related to EPZs of SMRs. The goal was to examine how the EPZ size might be scalable with respect to technological improvements and commensurate with the offsite consequences[17].

The study discussed the siting, source term for water and non-water-cooled reactors as well as consequences from multi-module accidents in terms of existing practices and strategies in Member States and existing IAEA safety requirements.

[16] IAEA, "Actions to Protect the Public in an Emergency due to Severe Conditions at a Light Water Reactor, EPR-NPP-PPA, Vienna (2013)
[17] IAEA SMR Regulators' Forum Pilot Project Report, "Report from Working Group on Emergency Planning Zone", (2018)

It was found that the IAEA safety requirements and the methodology are sufficient in their scopes and practices for determining the sizes of the EPZs (PAZ and UPZ). The study highlighted that the EPZ for SMRs can be limited by the site boundary due to:

- Small reactors and low power levels. Smaller reactor cores and lower power densities reduce the amount of potential radioactive material for potential releases to the environment. Also, the time response to an accident is less in SMRs since the water inventory is large compared to the power giving more time to execute actions in order to prevent or mitigate an accident. This means that the distances at which doses exceed accepted criteria could be lower.
- Modularity and multiple module facilities. In a multi-module plant solution, the source term is divided into smaller parts containing only a fraction of fuel compared to a large unit while still having a full set of safety systems in each module. This reduces the risk of large offsite releases.
- Improved containment functions. With compact, high-pressure resistant, multi-wall, below grade and water-immersed containment structures potential offsite consequences of SMRs will be lower.
- Separate operating and maintenance facilities. This pertains to SMRs without on-site refueling in which case the renewal of fuel (or whole reactor units) will take place in a dedicated facility.

Before the SMR Forum's work, already in 2010, the IAEA had concluded a Coordinated Research Project (CRP) on Small Reactors without On-Site Refuelling[18] during which a new concept of risk-informed methodology for EPZ sizing was proposed. The methodology can provide a definition of EPZ, once the basis acceptance criteria in terms of limiting dose and frequency have been provided (agreed upon with the regulator).

First, the proposed methodology would estimate the EPZ size for each reactor individually based on the safety performance of a specific plant design to account for innovations and enhancements made for new technologies. Second, the proposed methodology focuses on estimating the frequency of exceedance of the generic dose criteria (i.e., criteria defined by the regulator) for the full spectrum of accidents, i.e. on the risk. This differs from the current conservative practice where the EPZ size is largely defined by the deterministic analysis of the most severe design basis or beyond design basis accident and relies on estimation of frequency of an accident happening not dose limit exceeded.

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[18] IAEA, Small Reactors without On-site Refuelling: Neutronic Characteristics, Emergency Planning and Development Scenarios, IAEA-TECDOC-1652, IAEA, Vienna (2010).

To demonstrate the potential of the methodology, two case studies were carried out estimating EPZ size for 330 MWe (1000 MWth) IRIS-like reactors in Caorso, Italy and in Lithuania. In the case of Italy, the EPZ was reduced from 10 km (current US NRC prescribed distance) to less than 2 km and potentially to 1 km (if fuel handling effects would be reduced). This resulted in excluding two large towns of 180,000 inhabitants out of EPZ. In the Lithuanian case, it was studied how to provide heat and power after the closure of Ignalina NPP in 2009. It was found that while the EPZ size had no large effect on the electricity production potential, increasing the EPZ reduced the district heating potential.

Using this methodology in a performance-based licensing approach for improved or advanced plant designs would redefine the EPZ size but maintain the level of risk (level of dose and level of frequency) at an acceptable level. As a result, SMRs could be located closer to the users which carries economic benefits for the plant owner and societal benefits to the public as the needs for various demanding emergency planning activities and infrastructure elements is reduced.

A reverse application of this methodology could be used to determine the level of risk associated with currently set EPZ sizes for existing NPPs. This would indicate how much additional margin due to emotional perception of the nuclear risk is currently factored in.

European Atomic Energy Community (EURATOM)

The European Atomic Energy Community (EAEC or Euratom) is an international organization established by the Euratom Treaty on 25 March 1957 with the original purpose of creating a specialist market for nuclear power in Europe, by developing nuclear energy and distributing it to its member states while selling the surplus to non-member states.

EAEC shares the same Member States with the European Union but remains an independent body. This means that EURATOM is outside the regulatory control of the European Parliament.

Although the Euratom Treaty gives the Community no strict, exclusive powers in certain fields, it retains real added value for its members: The Commission has adopted recommendations and decisions which set European standards.

According to the Treaty, the specific tasks of EURATOM are:

- to promote research and ensure the dissemination of technical information
- to establish uniform safety standards to protect the health of workers and of the general public and ensure that they are applied
  Each Member State lays down the appropriate provisions, whether by legislation, regulation or administrative action, to ensure compliance with the basic standards of the Treaty. Each Member State is required to provide the Commission with the general data relating to any plan for the disposal of radioactive waste. At the same time, the assent of

the Commission is required where these plans are liable to affect the territories of other Member States
- to facilitate investment and ensure the establishment of the basic installations necessary for the development of nuclear energy in the EU
- to ensure that all users in the EU receive a regular and equitable supply of ores and nuclear fuels

Basic standards are laid down within the Community for the protection of the health of workers and the general public against the dangers arising from ionizing radiations. The expression 'basic standards' means:

1. maximum permissible doses compatible with adequate safety
2. maximum permissible levels of exposure and contamination
3. the fundamental principles governing the health surveillance of workers

These standards are expressed through directives issued by the Community:

- Council Directive 2014/87/EURATOM amending Directive 2009/71/EURATOM imposes obligations on the Member States to establish and maintain a national framework for nuclear safety.
- Council Directive 2013/59/Euratom establishes uniform basic safety standards for the protection of the health of individuals subject to exposures against the dangers arising from ionizing radiation.
- Council Directive 2011/70/Euratom imposes obligations on the Member States to establish and maintain a national framework for spent fuel and radioactive waste management.

As Estonia is part of the European Union and the Treaty, it is a Member State and must oblige to the directives laid down by the Community.

<u>Directive 2014/87/EURATOM amending Directive 2009/71/EURATOM</u>

Member States shall establish and maintain a national legislative, regulatory and organizational framework ("national framework") for the nuclear safety of nuclear installations. The national framework shall provide in particular for:

- the allocation of responsibilities and coordination between relevant state bodies
- national nuclear safety requirements, covering all stages of the lifecycle of nuclear installations
- a system of licensing and prohibition of operation of nuclear installations without a license
- a system of regulatory control of nuclear safety performed by the competent regulatory authority

- effective and proportionate enforcement actions, including, where appropriate, corrective action or suspension of operation and modification or revocation of a license

The determination on how national nuclear safety requirements referred to in the second point are adopted and through which instrument they are applied remains within the competences of the Member States.

The Member States also have to ensure that the license holder under the regulatory control of the competent regulatory authority, re-assesses the safety of the nuclear installation at least every 10 years.

The Directive also states that Member States shall ensure that their national nuclear safety frameworks require that nuclear installations are designed, sited, constructed, commissioned, operated and decommissioned with the objective of preventing accidents. Should an accident occur, the objective is to mitigate and avoid early radioactive releases that would require off-site emergency measures. This objective applies to nuclear installations for which a construction license is granted for the first time after 14 August 2014.

Directive 2013/59/EURATOM

This Directive establishes uniform basic safety standards for the protection of the health of individuals subject to occupational, medical and public exposures against the dangers arising from ionizing radiation.

By this Directive, Member States shall establish legal requirements and an appropriate regime of regulatory control which, for all exposure situations, reflect a system of radiation protection based on the principles of justification, optimization and dose limitation:

- Justification: Decisions introducing or altering an exposure pathway for existing and emergency exposure situations shall be justified in the sense that they should do more good than harm.
- Optimization: Radiation protection of individuals subject to public or occupational exposure shall be optimized with the aim of keeping the magnitude of individual doses, the likelihood of exposure and the number of individuals exposed as low as reasonably achievable taking into account the current state of technical knowledge and economic and societal factors.
- Dose limitation: In planned exposure situations, the sum of doses to an individual shall not exceed the dose limits laid down for occupational exposure or public exposure. Dose limits shall not apply to medical exposures.

Public exposure 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.

The Directive states that the limit on the effective dose for occupational exposure is 20 mSv in any single year. However, a higher effective dose of up to 50 mSv may be authorized by the competent authority in a single year, provided that the average annual dose over any five consecutive years does not exceed 20 mSv.

National regulators

United States Nuclear Regulatory Commission

The NRC was founded by the Energy Reorganization act in 1974. Its main role is licensing and regulating the use of radioactive materials for civilian purposes while protecting people and the environment. The NRC regulates commercial nuclear power plants and other nuclear applications such as the production of radioactive isotopes for medical uses. The NRC does not own nor operate nuclear power plants.

The main scopes that the NRC work on can be summarized in the following list[19]:

- Setting rules for radioactive materials use, while these rules protect the workers and the public from any potential hazard, noticing that the NRC takes into consideration the views of the public, industry representatives, researchers, state officials, scientists and technical experts
- Licensing a nuclear facility that involves the use of radioactive materials, where the type and quantities of these radioactive materials must be specified.
- Inspection of a licensed nuclear facility in order to ensure that regulations and terms meet the requirements.
- The NRC has the right to enforce regulations if a violation takes place, and in that case the license can be suspended.
- Continuous evaluation of nuclear reactors and radioactive materials facilities, where the NRC can address crucial weaknesses in the design, operations, procedures and equipment.
- Regulatory research provides technical advice, analytical tools and information to support NRC decisions, focusing on safety and security.
- The NRC provides a program to ensure the readiness and response to an accident at a nuclear facility that may lead to potential hazard if not mitigated.

[19] US NRC website, https://www.nrc.gov, (accessed 2019).

Regarding nuclear facilities with SMRs and other new technologies (ONT), the NRC has identified that there are differences between SMRs/ONTs and existing large-reactor-based fleet including unique aspects of new designs, the applicability of current regulatory requirements and lack of international experience with licensing advanced reactor designs.

In 2010, the NRC staff has stated the following on the need for change of regulation of SMRs and other advanced reactors:

"*To provide for more timely and effective regulation of advanced reactors, the Commission encourages the earliest possible interaction of applicants, vendors, other government agencies, and the NRC to provide for early identification of regulatory requirements for advanced reactors and to provide all interested parties, including the public, with a timely, independent assessment of the safety and security characteristics of advanced reactor designs. Such licensing interaction and guidance early in the design process will contribute towards minimizing complexity and adding stability and predictability in the licensing and regulation of advanced reactors.*"

The NRC identified potential policy and licensing issues resulting from i) the key differences between the new designs and current-generation LLWRs (such as size, moderator, coolant, fuel design, and projected operational parameters) and also ii) from industry-proposed review approaches and industry-proposed modifications to current policies and practices. The issues currently open are[20]:

- Appropriate Source Term, Dose Calculations, and Siting for SMRs
- Offsite Emergency Planning Requirements for SMRs and other new technology
- Insurance and Liability for SMRs
- Security and Safeguards Requirements for SMRs

The two topics that are especially relevant to reactor safety and siting (i.e., the topic of this study) are the source term and offsite emergency planning. Among the national regulators described here, the US NRC is the one using a rather prescriptive approach based on a framework that has been optimized over time in order to efficiently license large LWRs. Since SMRs explore a very different paradigm than large reactors the prescriptive, it is the LLWR optimized aspects of the existing framework that require the most changes in order to accommodate new trends.

Emergency Planning Zones

Existing regulation (10 CFR 50.47) developed for large LWRs require a 10-mile plume exposure pathway EPZ and 50-mile ingestion exposure pathway EPZ to prevent food and water contamination (see Figure 5). While somewhat cumbersome, it has been possible to request an

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[20] https://www.nrc.gov/reactors/new-reactors/smr.html Accessed: 2020-12-18

exemption to these requirements for small LWRs and non-LWRs e.g., in the 1980 Final Rule, the NRC clarified that the size of the EPZ could be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MWth.

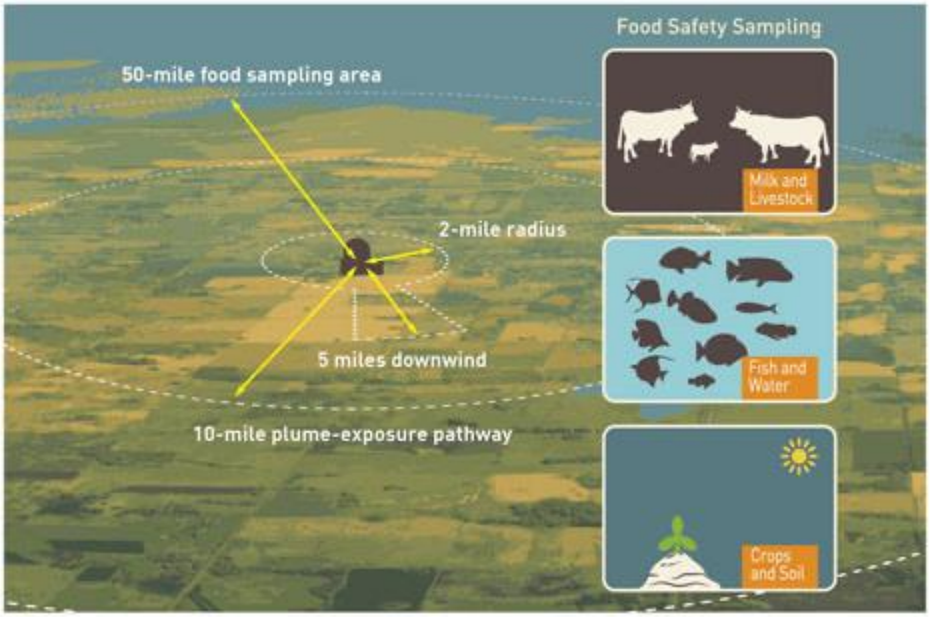

Figure 5: The 10-mile plume pathway EPZ and 50-mile ingestion EPZs. The 2-mile ring around the plant together with the 5-mile zone in the downwind of projected release are identified for the evacuation.

Based on a topical white paper prepared by NEI in 2013[21] and following several rounds of discussions the NRC proposed[22] and issued a draft regulatory guide DG-1350[23] in May 2020 to amend its regulations to include new alternative EP requirements for SMRs and other new technologies, such as non-LWRs and certain non-power production or utilization facilities (NPUFs).

The new alternative EP requirements and implementing guidance in DG-1350 would adopt a performance-based, technology-inclusive, risk-informed, and consequence-oriented approach. The new alternative EP requirements would adopt a scalable plume exposure pathway EPZ approach and address ingestion response planning. The new rules would:

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[21] NEI, „Proposed Methodology and Criteria for Establishing the Technical Basis for Small Modular Reactor Emergency Planning Zone“, White Paper, 2013 (web: https://www.nrc.gov/docs/ML1336/ML13364A345.pdf)
[22] NRC, "Emergency Preparedness for Small Modular Reactors and Other New Technologies," Federal Register, Vol. 85, No. 92, May 12, 2020 (web: https://www.federalregister.gov/documents/2020/05/12/2020-09666/emergency-preparedness-for-small-modular-reactors-and-other-new-technologies)
[23] NRC, "Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities," Draft regulatory guide DG-1350, May 2020

1. continue to provide reasonable assurance that adequate protective measures can and will be implemented by an SMR or ONT licensee
2. promote regulatory stability, predictability, and clarity
3. reduce requests for exemptions from EP requirements
4. recognize advances in design and technological advancements embedded in design features
5. credit safety enhancements in evolutionary and passive systems
6. credit smaller sized reactors' and non-LWRs' potential benefits associated with postulated accidents, including slower transient response times, and relatively small and slow release of fission products

According to the newly proposed approach the applicants and licensees of SMRs utilities would have the option to develop a performance-based EP program as an alternative to using the existing, deterministic EP requirements in 10 CFR part 50. This means the planning and EPZ sizing would be determined case-by-case based upon the knowledge of potential consequences, timing, and radiological release characteristics from a spectrum of accidents. Emergency preparedness is risk-informed rather than risk-based, and therefore emergency planning is independent of accident probability but depends rather on dose/distance effects.

The principle of using dose savings to determine EPZ size has been used in the past when the NRC licensed several small reactors with a reduced EPZ size of 5 miles (8 km). In the new approach, applicants would need to establish a plume exposure EPZ such that public dose does not exceed 10 mSv TEDE over 96 hours from the release.

The proposal would allow SMRs to have the EPZ at the site's boundary, which would exempt operators from offsite radiological emergency planning and the Federal Emergency Management Agency (FEMA) from evaluation of the site's emergency plans.

Even if the EPZ is bounded by the site boundary the applicant would still need to reference capabilities of Federal, Tribal, State, and local authorities since they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.

The NRC staff is scheduled to provide the final rule to the Commission for approval on September 30, 2021.

Source term

Source term contributes to a number of licensing areas including, for example, siting, control room habitability, emergency preparedness, and security considerations.

Source term determination can be divided in two main categories:

- Category 1: An approach assuming that the SMR design features or event progression are similar to the conventional large light water reactors (LLWRs). In this case the SMR source term calculation methodology follows the currently used methodology for LLWRs.
- Category 2: An approach that takes into consideration the unique features of the SMR designs, called a mechanistic source term (MST) methodology.

Source term of Category 1 will be used to analyze offsite and control room doses for SMR designs that are similar to the current LLWRs, and the SMR vendors will follow the regulatory guide RG 1.183, which this a guide set by the NRC in order to provide guidance to licensees of operating power reactors on acceptable applications of alternative source term methodology (AST)[24].

However, the main objective of SMRs design is to create nuclear power plants with simplified operational and enhanced safety features that are significantly different from the conventional LLWRs. The need for another methodology source term methodology, denoted as the Category 2, arises from considerations of unique SMR features. An example of the need to have this category is that large break LOCA cannot be a postulated accident to determine the source term in most SMRs as they avoid using large diameters pipes, eliminating the accident by design.

SMRs designers work on forming one or more surrogate accident scenarios, which will be denoted as source term design basis accidents (STDBA) that will meet the regulator intention to address the maximum hypothetical accident (MHA). MHA is by definition an accident that can lead to a core meltdown and result in a radioactive release.

The SMR designers have three options to address the MHA, the first one to follow the RG 1.183 core damage scenario for the small break LOCA, however, this will be extremely conservative and ignores the SMRs state of art when it comes to their safety features.

The second, MST approach is based on the reactor-specific analysis of fission product release due to fuel, cladding and core damage. This results in reactor-specific accident sequences being evaluated. The MST relies upon the best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environment. The MST approach takes into consideration the differences between SMR designs and the current licensed LWRs.

A third option is a hybrid one based on the RG 1.183 source term but modifying some aspects to take into account the differences in SMR transient response behavior such as the timing of the onset fuel damage. Noting that both hybrid and MST methodologies will require the development

[24] United States Nuclear Regulatory Commission, regulatory guide 1.183 "Alternative radiological source terms for evaluation design basis accidents at nuclear power reactors", July, 2000.

of an SMR PRA to identify the events that can lead to core damage. Table 3 summarizes the three approaches[25]:

Table 3: Comparison of STDBA source term methodologies.

Methodology	Advantages	Disadvantages
R.G 1.183 Source term	• Clear regulator guidance specification • Technically simplest	• No credit for SMR source term regarding design features. • Highest expected source term
Mechanistic Source Term (MST)	• Takes credit for SMR design features to reduce both magnitude and timing of source term • Lowest source term, but still conservative	• Technically most complex • Lowest regulator certainty
Hybrid of RG 1.183 and MST Source Term	• Some SMR design performance incorporated to increase timing • Less complex than MST • Better regulator certainty than MST	• Does not credit all source term design features improvements

After multiple interactions with the U.S. Department of Energy (DOE) and nuclear industry organizations in 2016 the NRC staff concluded that SMR applicants can employ modern analysis tools to demonstrate quantitatively the safety features of those designs, and MST analysis methods can also be used by applicants to demonstrate the ability of the enhanced safety features of plant designs to mitigate accident releases, allow future license applicants to consider reduced distances to Exclusion Area Boundaries and Low Population Zones and potentially reduce distance to population centers.

In May 8, 2020, the staff provided options and a recommendation (SECY-20-0045[26]) to the Commission on possible changes to regulatory guidance to address population-related siting considerations for SMRs. The staff's recommendation is to pursue a revision to the population-related siting guidance to provide technology-inclusive, risk-informed, and performance-based

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[25]Nuclear Energy Institute, "Small modular reactor source terms", NEI, December 27, 2012 (web: https://www.nrc.gov/docs/ML1336/ML13364A345.pdf)
[26] US NRC, "Population Related Siting Considerations for Advanced Reactors", SECY-20-0045, May 8, 2020 (web: https://www.nrc.gov/docs/ML1914/ML19143A194.pdf)

criteria to assess certain population-related issues in siting SMRs. The process is ongoing, and the staff is waiting for Commission's directions.

Canadian Nuclear Safety Commission

The Canadian Nuclear Safety Commission (CNSC) is Canada's nuclear regulator and operates under the authority of the Nuclear Safety and Control Act (NSCA). The CNSC regulates the use of nuclear energy and materials to protect health, safety, security and the environment; to implement Canada's international commitments on the peaceful use of nuclear energy; and to disseminate objective scientific, technical, and regulatory information to the public.

The CNSC regulates using a risk-informed approach, which is long-established and forms the foundation of its regulatory activities. The CNSC sets requirements and provides guidance on how to meet them, and the applicant or licensee may put forward a case to demonstrate that the intent of a requirement is addressed by other means. Such a case must be demonstrated with suitable supporting evidence.

CNSC's licensing process (Figure 6) provides for significant flexibility. The EA and various license applications can be reviewed in parallel, or in series. The Commission may also consider applications for combinations of activities; for example, a license to prepare site and construct or license to construct and operate, as long as the proponent addresses all requirements associated with the proposed activities.

Figure 6: Process for each licensing phase in facility lifecycle.

The Canadian nuclear regulatory framework is comprehensive and in large part technology neutral, which means that it allows for all types of technologies to be safely regulated. This means that in regulating SMRs, the CNSC can apply the same criteria used to regulate traditional reactor facilities. This will be done through a risk-informed approach, by applying resources and regulatory oversight commensurate with the risk associated with the regulated activity[27].

For edge-of-grid or even off-grid applications in remote parts of Canada, the CNSC is aware that several vendors are considering SMR concepts in the 3 to 35 MWe (per unit) range. They are being considered by vendors as either supplementary to an existing northern grid-system or as an off-grid source.

On EPZ

In Canada, there are two primary types of planning zones (as depicted in Figure 7):

Exclusion zone: Per section 1 of the Class I Nuclear Facilities Regulations, an "exclusion zone" is a parcel of land within or surrounding a nuclear facility on which no permanent dwellings are allowed and over which a licensee has the legal authority to exercise control.

Emergency planning zone: An emergency planning zone (EPZ) is defined as the area in which implementation of operational and protective actions might be required during a nuclear emergency, to protect public health, safety, and the environment. An EPZ addresses emergency measures to be used outside the licensee's exclusion zone and that are normally controlled and executed by an external emergency planning authority.

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[27] CNSC, "Small Modular Reactors: Regulatory Strategy, Approaches and Challenges," 2016.

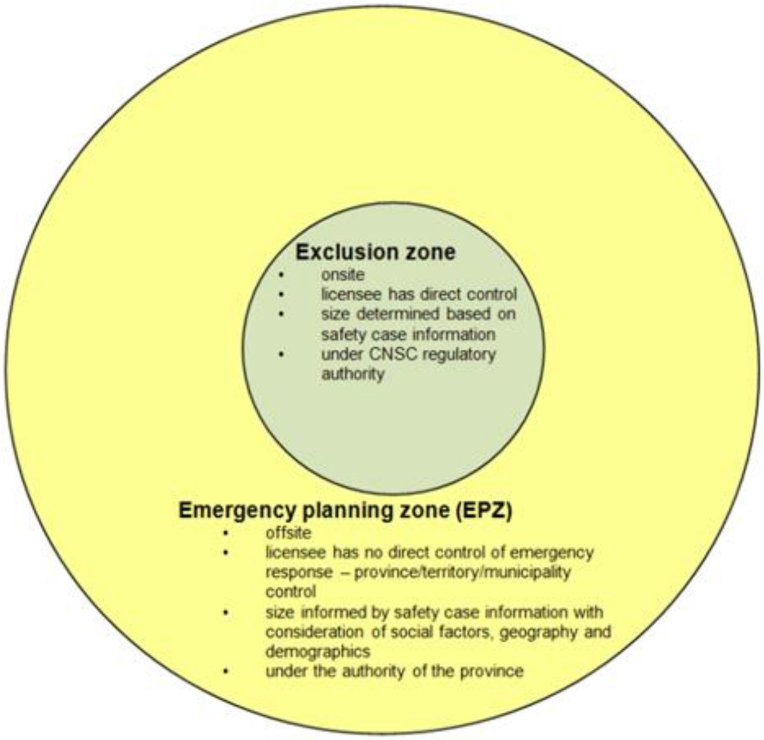

Figure 7: Exclusion zone and emergency planning zone according to CNSC.

There are no legislative or regulatory requirements for EPZ sizing in Canada and therefore no restrictions currently in place on minimum EPZ size. EPZ and other planning actions should be undertaken in relation to the risks associated with the specific technology. As such, results from safety analyses (i.e., the probabilistic safety analysis) in combination with the protection strategy used by offsite planners will determine the EPZ size. This is consistent with the overall methodologies documented by the IAEA.

In Canada, different agencies and participants have different responsibilities:

- **Provinces/Territories**. Provincial and territorial governments have primary responsibility for offsite emergency planning and response to protect public health, property and the environment. Each province prepares its provincial nuclear emergency response plan (PNERP) in coordination with the federal government, under the Federal Nuclear Emergency Plan (FNEP).
- **Health Canada**. Health Canada, as the lead department under the FNEP (Health Canada, 2002), provides guidelines for intervention following a nuclear emergency in Canada or affecting Canadians. These guidelines are a key reference for provincial governments when preparing provincial nuclear emergency plans.
- **Canadian Nuclear Safety Commission**. The CNSC considers the design-basis accident dose limits and confirms that the determined exclusion zone distance is appropriate to meet all safety requirements. The CNSC works closely with the province to provide information

about the nuclear facility's safety case and licensing process to help the province determine the EPZ.

- **Applicants for activities involving new reactor facilities**. Applicants and licensees for activities involving the use of reactor facilities are responsible for submitting complete applications outlining how the site evaluation and chosen technology will, through their safety analysis, result in an appropriate exclusion zone and emergency response plans to meet provincial requirements.

Although the determination of the EPZ size is under the province's authority, the province works with multiple supporting organizations to develop a technical planning basis that would be used to determine the EPZ. In summary, the EPZ extent is based on the nuclear reactor's technology, the resulting dose assessments against the provincial PALs, and various external factors such as social considerations, demographics and geography. Figure 8 shows the process for defining an EPZ.

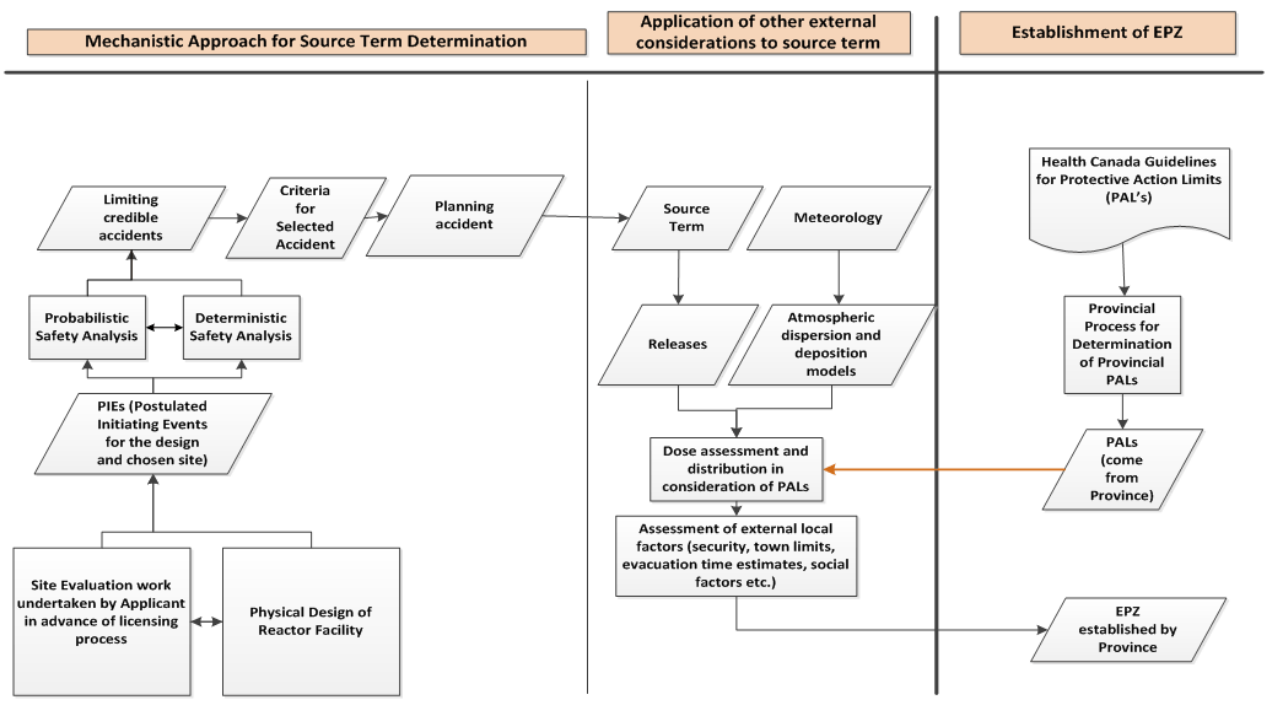

Figure 8: An overview of the Canadian process for defining the EPZ.

During the whole process, all of the aforementioned parties are involved. In addition, the provincial authorities would also consider social factors, geography and demographics in determining the EPZ around the nuclear facility.

<u>The Office for Nuclear Regulation</u>

The Office for Nuclear Regulation (ONR) is responsible for the independent regulation of nuclear safety and security across the United Kingdom (UK). ONR currently regulates 36 nuclear licensed sites in the UK[28].

The UK government launched an initiative in December 2017 for developing advanced reactors. This is formally known as the AMR Feasibility & Development (F&D) project. As part of this project, the Department for Business, Energy and Industrial Strategy (BEIS) requested advice from ONR on the level of confidence in seven Advanced Modular (AMR) fission reactor designs being able to meet UK regulatory requirements in the future. In May 2019, ONR delivered the advice to BEIS following evaluation of the submissions provided by the seven vendors (incl. sodium fast reactor by ARC; three gas cooled reactors by DBD, USNC and U-Battery; molten salt reactor by Moltex Energy; and two lead fast reactors by Westinghouse and LeadCold).

In 2019/2020, ONR modernized the Generic Design Assessment (GDA) process to enhance the efficiency and flexibility of the process, taking account of learning from previous assessments, the government's Nuclear Sector Deal and the potential for Advanced Nuclear Technologies (ANTs) to enter GDA.

The framework for the protection of members of the public and workers from and in the event of radiation emergencies are set out in the Radiation (Emergency Preparedness and Public Information) Regulations 2019 (REPPIR)[29]. In case there is a potential for off-site release of radioactivity within the UK that would require implementation of countermeasures, EPZs are designated.

The ONR defines two types of emergency planning areas:

- Detailed Emergency Planning Zone (DEPZ) - a zone around a facility for which the REPPIR requires the local authority to prepare a detailed off-site emergency plan with the purpose of restricting public exposure in the event of a reasonably foreseeable radiation emergency.
- Outline Planning Zone (OPZ) - an area beyond the DEPZ. The presence of an OPZ assists local authorities in planning for extremely unlikely but more severe events.

The size of the emergency planning area differs site by site in the UK, with due consideration given to individual factors associated with each site. Following the publication of ONR's revised

[28] ONR website, http://www.onr.org.uk/aims-and-objectives.htm, (Accessed Aug 10, 2020)
[29] ONR, "Approved Code of Practice and Guidance For The Radiation (Emergency Preparedness and Public Information) Regulations 2019", REPPIR, (2019)

principles (January 2014), ONR commenced revision of the offsite emergency planning areas to defined maps.

Similar to the new approach proposed by the NRC in the US, determining the DEPZ in the UK is already risk-informed not risk-based. Evaluating the likelihood of potential radiation emergencies is important in relation to ensuring a proportionate overall approach to emergency planning. Initiating events leading to fault sequences protected by the same safety systems and equipment, and resulting in similar consequences, should be grouped, and their associated sequence frequencies summed. The source term selected to represent the group of sequences should be the most limiting one in terms of the radiological dose. Impact (Table 4) and likelihood (Table 5) data is used in the REPPIR risk framework (Figure 9), which determines the need for the planning zones.

Table 4: Impact table.

Descriptors						
	Impact descriptor and effective dose	**Human life**	**Health & Safety**	**Quality of Life**	**Property**	**Environment**
A	Catastrophic (> 1 Sv)	Death and life changing consequences severe deterministic effects possible.	Possibility of life changing consequences because of significant (> 5%) increased risk of cancer induction.	Complete reconstruction of life activities.	Asset value completely lost.	Exclusion zones increase and heavy restrictions extended to further distance.
B	Significant (100-1000 mSv)	Possibility of moderate deterministic effects.	Possibility of life changing consequences because of very small (0.5%) increased risk of cancer induction.	Initial reconstruction and continued interruption of normal life activities.	Major asset value depreciation.	Exclusion zones of environmental areas and heavy restrictions.
C	Moderate (10-100 mSv)	No potential for deterministic effects, below threshold dose.	Possibility of life changing consequences because of very small (0.5%) increased risk of cancer induction.	Enforced prevention or interruption of normal life activities.	Potential or real asset value depreciation.	Restricted or temporary loss of environmental growth or produce.

D	Minor (1-10 mSv)	No potential for deterministic effects, below threshold dose	Minimal impacts and unlikely to have life changing consequences	Potential self-imposed restrictive changes in normal life activities.	Assumed asset value depreciation.	Reluctance to use environmental areas and produce.
E	Limited (< 1 mSv)	No potential for deterministic effects, below threshold dose.	Normal background	Sustained normal life activities.	Asset value sustainable or dominated by market forces.	Sustained environmental conditions.

Table 5: Likelihood table.

Likelihood descriptor	**Relative likelihood of occurring in the next 5 years**
Events not considered in the design	Less than 1 in 20,000
Very low	1 in 20,000 – 1 in 2,000
Low	1 in 2,000 – 1 in 200
Medium	1 in 200 – 1 in 20
High	1 in 20 – 1 in 2
Very high	Greater than 1 in 2

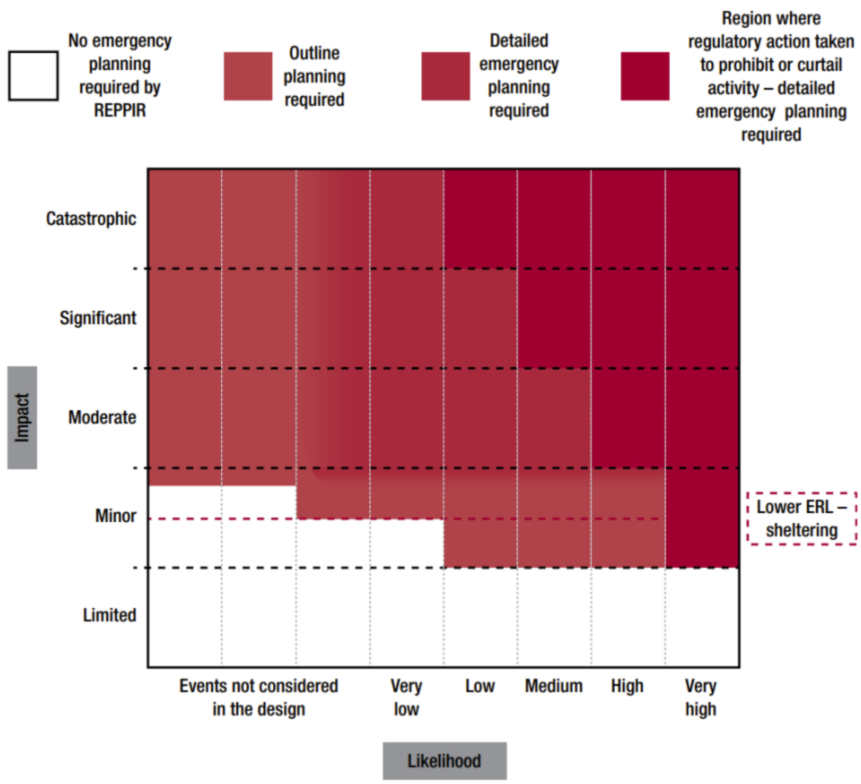

Figure 9: REPPIR risk framework.

By the REPPIR Risk framework, off-site emergency planning is not needed if the off-site effective dose is less than 1 mSv (Limited impact in Figure 9). Also, in case of a possible minor impact (1-10 mSv), off-site emergency planning would not be necessary in case of a very low likelihood.

Only OPZ would be required in case of a minor impact and up to high likelihood. For the operators of smaller, lower-risk premises this means the possibility not to purchase meteorological data and perform significant amounts of atmospheric dispersion modelling as part of the requirements of a full consequence assessment.

5. Reactors

NuScale (NuScale Power LLC)

The SMR solution developed by NuScale Power LLC is based on 77 MWe NuScale Power Modules (NPMs). Multiple (4-12 module solutions considered) NPMs can be installed to form a large power plant to supply energy for electricity, district heat, and a variety of process heat systems. The modules are fully fabricated in factories and transportable by rail, truck or ship to the site for assembly and installation with other operating modules. Factory production aims to improve component quality and reduce cost, enabling shorter plant construction schedule.

The NPM is based on an integral pressurized water reactor design containing the reactor pressure vessel (RPV) and the containment vessel (Figure 10). The 18.8 m x 3 m steel RPV encases the major nuclear components such as the reactor coolant system, pressurizer, core, riser and a steam generator. The containment vessel is a secondary steel pressure vessel housing the entire RPV and control rod drive mechanisms, sensors, valves, and associated piping as well as the Decay Heat Removal System (DHRS) and Emergency Core Cooling System (ECCS).

All NPMs are submerged in a common steel-lined concrete-walled reactor pool filled with water to provide long term core cooling in case of a severe accident. The reactor pool is capable of absorbing all decay heat generated by the 12 modules for more than 30 days followed by air cooling for an unlimited length of time. The NPM and the reactor pool are below ground. See the 12-module plant layout in Figure 11.

The improved safety of the NuScale plant compared to existing plants is achieved through design simplifications, passive operational and safety features and small fuel inventory. Passive core cooling is achieved through natural circulation in both, normal operation and accident situations. Heat is transferred from the reactor core, through the helical-coil steam generator to the secondary coolant due to buoyancy-driven flow. The integral design and passive core cooling by natural circulation eliminate the need for pumps and reduce the amount of necessary piping and valves which, in turn, eliminates large break LOCA scenarios. Such a technological approach would also reduce the need for maintenance and potential failures associated with those components. Use of passive safety systems eliminates the need for external power in accident conditions. Passive safety systems include the reactor coolant system (including reactor safety valves), the DHRS, ECCS, containment isolation system and control room habitability.

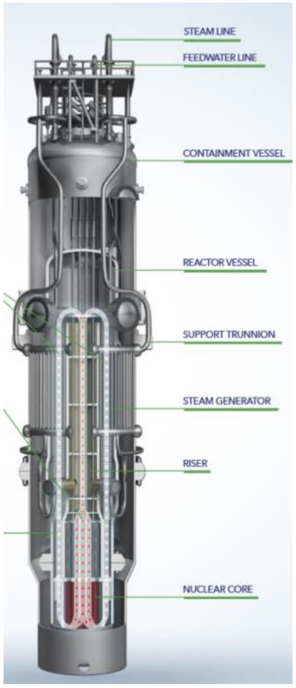

Figure 10: The integrated NuScale Power Module.

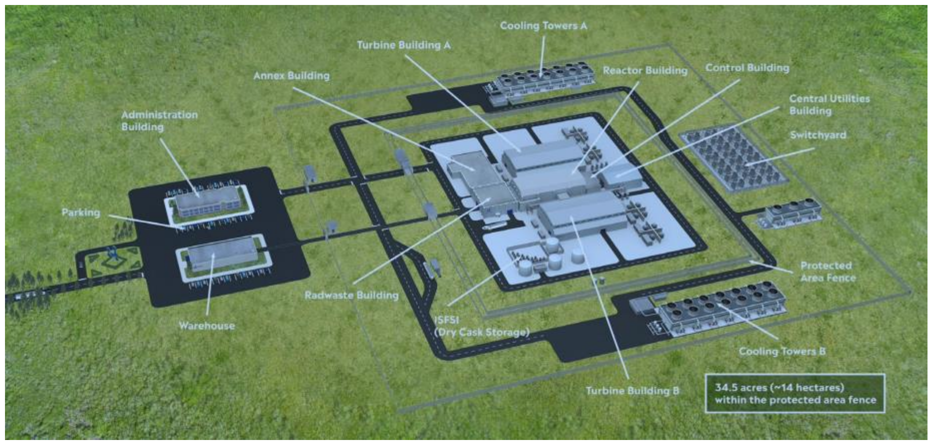

Figure 11: Layout of a standard 12-module NuScale plant encompassing and are of 0.14 km^2.

The two fully independent and redundant safety systems, DHRS and ECCS, are implemented in each module (Figure 12). ECCS is activated when the steam generator is not available as a heat sink to remove the heat from the primary system. ECCS also ventilates the steam out from the RPV to the containment where it condenses on the outer walls in contact with the water pool. The DHRS provides cooling to the secondary side of the reactor in case of non-LOCA events when feedwater is not available. The decay heat is removed by natural circulation in two redundant trains, each connecting a steam generator to a condenser immersed in the reactor pool and capable of removing 100% of the decay heat.

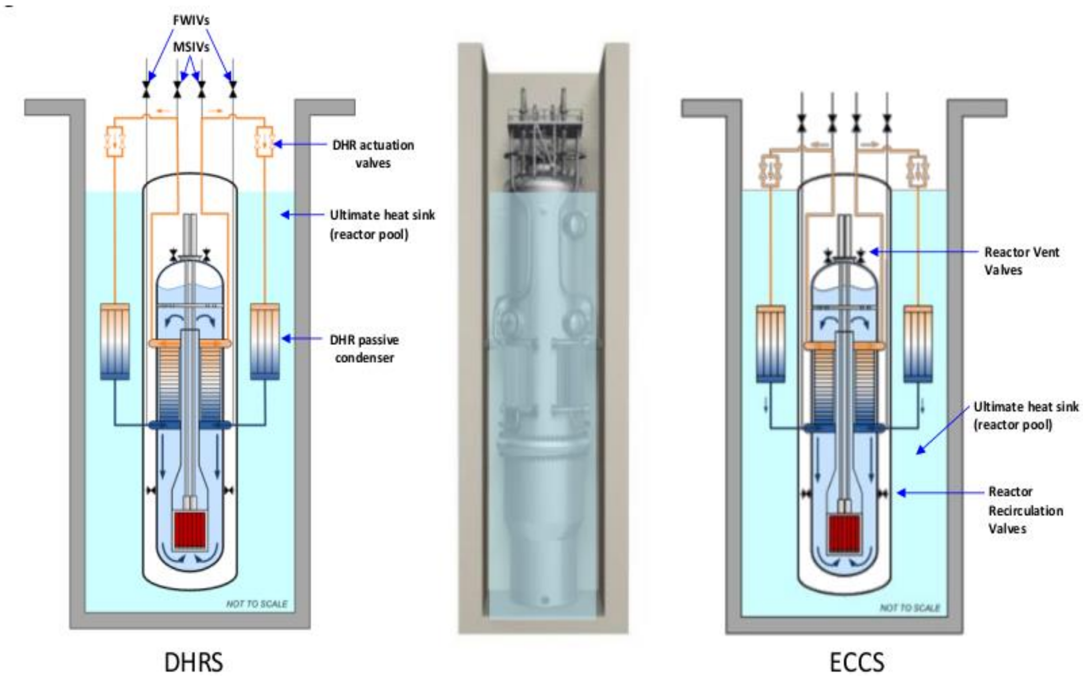

Figure 12: Diagrams of the DHRS (left) and ECCS (right) in comparison with a solid body model of an NPM (middle).

The NuScale design effort has been supported by a one-third scale, electrically heated integral test facility that operates at full pressure and temperature. Operation of such a test facility has contributed to continuous design optimization, learning and validation of safety performance. Mechanical and thermal-hydraulic testing of the fuel, cladding and structural materials of the fuel assemblies has been completed in Framatome's test facilities.

As the NuScale design relies on well-established LWR technology, it can be licensed within the existing LWR regulatory framework, drawing on a vast body of established research and development, proven codes and methods, and existing regulatory standards.

NuScale Power, LLC submitted an application for standard plant design certification to the NRC in December 2016. Less than 4 years later in August 2020 the NRC completed the final safety evaluation report (FSER) and issued a Standard Design Approval (SDA) confirming that the plant design meets the applicable requirements for the design certification stage of licensing[30,31]. NuScale aims to receive a full design certification in October 2021. The company is also engaged in the three-phase pre-licensing Vendor Design Review (VDR) process at CNSC in Canada. Multiple submittals have been made to the VDR Phase 2 and the process is expected to continue through 2021.

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30 https://www.nrc.gov/docs/ML2023/ML20231A804.pdf
31 https://www.nrc.gov/reading-rm/doc-collections/news/2020/20-043.pdf

Emergency Planning Zone

Methods to determine EPZ sizing vary by country. US NRC regulations currently set EPZs at a fixed distance around nuclear power plants for the entire country while in Canada, the province in which the nuclear power plant is located determines the EPZ based on the safety case and local factors. Since the process in Canada is still in the pre-licensing phase, we focus on the developments in the US.

To account for advancements in the design and avoid the pre-determined EPZs in the future siting/construction/operating licensing processes, NuScale submitted a licensing topical report (LTR) proposing a methodology for design-specific plume exposure pathway (PEP) EPZ determination for the NuScale SMR plant design[32]. The ingestion EPZ is not addressed, as the determination of this distance depends mostly on the site-specific considerations.

The design-specific methodology NuScale proposing is an extension of the approach presented by the Nuclear Energy Institute (NEI) for the risk-informed EPZ methodology[33]. Namely, NuScale has extended the NEI methodology regarding the appropriate accident sequences to be included and consider a consequence-oriented approach i.e., the one estimating the risk of dose exceedance per distance from the plant.

According to the methodology, there are 3 dose-based criteria a future applicant shall adhere to:

- Criterion a: The EPZ should encompass areas in which projected dose from design basis accidents (DBAs) could exceed 10 to 50 mSv TEDE (requiring evacuation and sheltering, respectively).
- Criterion b: The EPZ should encompass areas in which consequences of less severe accidents could exceed 10 to 50 mSv TEDE (requiring evacuation and sheltering, respectively).
- Criterion c: The EPZ should be of sufficient size to provide for substantial reduction in early severe health effects in the event of more severe accidents (a threshold of 2 Sv TEDE).

Based on the Final Safety Analysis Report (FSAR) in the Design Certification Application, NuScale has no design basis accidents that would lead to core damage[34]. Therefore, the DBAs have to meet the Criterion a. For the Criterion b and c, the core damage source term (CDST) is

[32] NuScale Power, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones," TR-0915-17772-NP, Revision 2, August 2020. ADAMS Accession Number ML20217L422.
[33] https://www.nrc.gov/docs/ML1336/ML13364A345.pdf
[34] NuScale Power, "Chapter Fifteen: Transient and Accident Analyses," NuScale Standard Plant Design Certification Application, Part 2, Tier 2, Revision 5, July 2020. ADAMS Accession Number ML20224A504.

evaluated using integrated deterministic safety analysis and probabilistic risk assessment of a postulated surrogate core damage event (CDE). The surrogate CDE CDST is based on the median release fractions, shortest time to release, and shortest release duration, from a suite of five BDBA CDSTs. The overall combined frequency of the CDST has been estimated to be $5x10^{-11}$ per year.

The reactor's simplicity of design, reliable passive safety features, small source term, multiple fission product barriers, and the independence of each module, contribute to having the EAB, LPZ (areas considered during site evaluation) and EPZ at the site boundary of 0.5 km radius.

If a credible initiating event was determined to be able to impact multiple modules (e.g., a loss of AC power to all modules), the number of modules impacted would be determined, and a total source term to the environment would be calculated (see Section 3.4.4 of the NuScale EPZ methodology).

It should be noted that the NRC during its thorough review of the NuScale design did not find any credible event that would impact multiple modules and cause a release from the facility that would require an EPZ extending beyond the site boundary.

BWRX-300 (General Electric-Hitachi)

GE-Hitachi Nuclear Energy's BWRX-300 is a 300 MWe small modular boiling water reactor that is cooled by natural circulation during normal and abnormal operations. The design is the 10th generation of GE's (predecessor of GEH) BWR concept since 1955 (Figure 13) and an evolution of the 1520 MWe Economic Simplified Boiling Water Reactor (ESBWR) already licensed by the US NRC. The BWRX-300 is designed for base load electricity generation and district heating with a load following capability in the range of 50 to 100% at a ramp rate up to 0.5% per minute, and a target capacity factor is 95%. The reactor is optimized and simplified to minimize the cost while maintaining the required safety to make it competitive with the natural gas-fired plants.

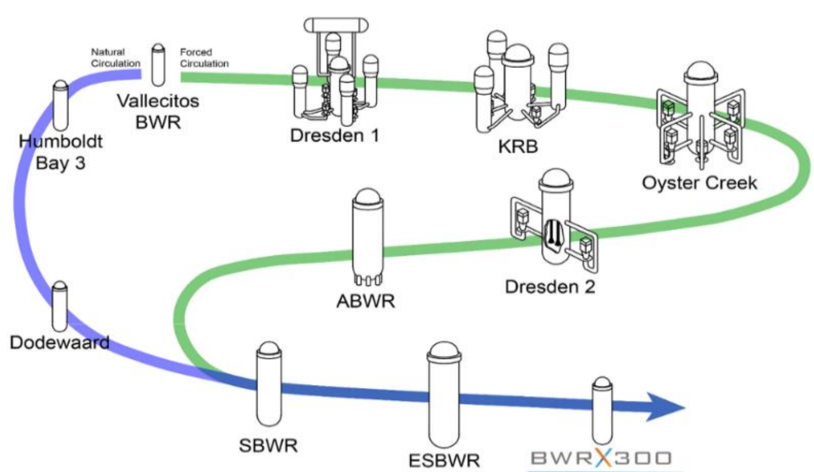

Figure 13: Historical evolution of BWRs to BWRX-300.

Approximately five times less powerful, the BWRX-300 is a down-scaled and more integrated interpretation of the ESBWR design principles (Figure 14). The majority of the internal components are designed to be removable when the reactor pressure vessel (RPV) is opened for refueling or maintenance. The RPV is approximately 4 m in diameter and 27 m in height, and this height and resulting distance between heat source and sink supports efficient heat removal using natural circulation. This RPV is located inside a dry primary containment vessel (PCV) located mostly below grade, which consists of a leak-tight nitrogen inert gas space designed to confine radioactive fission products, steam and water released in the unlikely event of LOCA. Large-break LOCAs are isolated using RPV isolation valves. The reference site of BWRX-300 is confined in a 260 m x 332 m footprint including the power plant, switchyards, cooling tower (if needed) and support facilities (Figure 15).

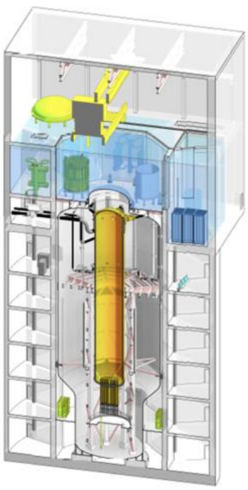

Figure 14: BWRX-300 reactor building.

Figure 15: Conceptual site layout with two BWRX-300 units.

BWRX-300 technical design features that are new to the BWR technology include:

1. RPV isolation valves help mitigate the effects of a LOCA. All pipe systems greater than 50 mm in diameter have double isolation valves.
2. Safety relief valves (SRVs) are eliminated since they are historically the most likely causes of LOCAs.
3. Dry containment which has been proven to effectively contain the releases of steam, water and fission products after a LOCA.
4. Designed to minimize the costs of construction, operation, maintenance, staffing and decommissioning. This includes simplified design, fewer safety related systems, components and water pools, use of natural circulation and below grade reactors.

5. Use of off-the-shelf GEH power conversion components (generators, turbines) that are operated in many power plants worldwide is possible due to the smaller reactor size.

The BWRX-300 safety design philosophy is built on utilization of inherent margins (e.g., larger structure volumes and water inventory) to mitigate system challenges. Off-site power is not a safety requirement, and its availability should not be included as a requirement when licensing the specific plant. Increased capacity active systems are used for the feedwater pumps and control rod drive mechanisms; however, passive systems are used as another line of defense to provide confidence in the plant's ability to handle transients and accidents.

The decay heat removal after any reactor isolation situation is achieved using an isolation condenser system (ICS). It consists of three independent heat exchanger loops with a capacity of approximately 33 MWth. The system is initiated automatically under abnormal conditions, and if a loss of DC power occurs. It can also be initiated manually by the operator from the main control room by opening the IC condensate return valve. Steam generated in the RPV is guided to the isolation condenser (IC) system where it condenses on the tube side and transfers heat to the water in the IC pool where it is vented to the atmosphere. Such heat transfer process is accomplished by natural convection i.e., no pumping equipment is required. The heat rejection process can be continued beyond seven days by replenishing the pool inventory. See the IC system in Figure 16.

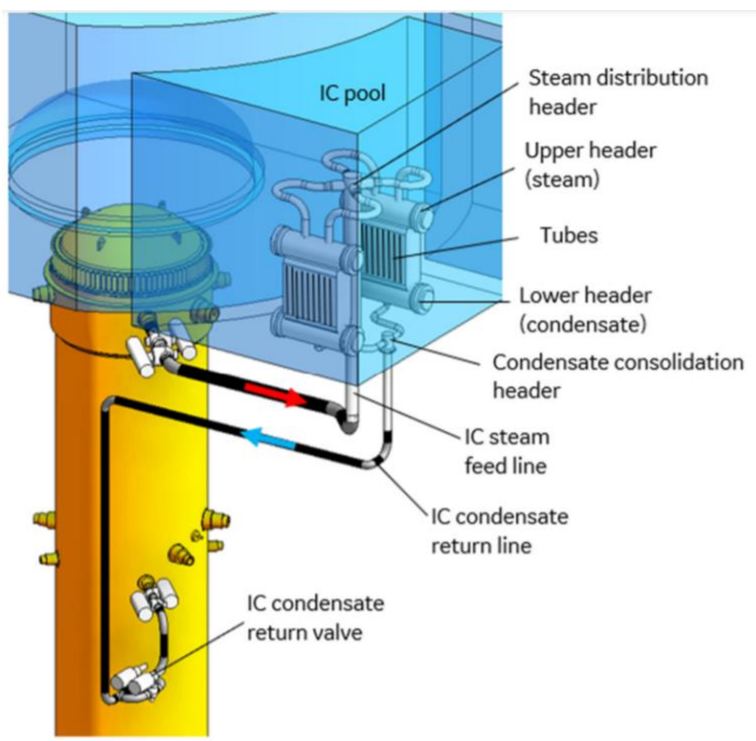

Figure 16: BWRX-300 isolation condenser system (ICS).

A passive containment cooling system (PCCS) consisting of several low-pressure heat exchangers is used to remove the heat from the PCV and transfer it to a water pool above it. From the pool the heat is vented to the atmosphere. PCCS requires no sensing, control, logic or power-actuated devices for operation. Since there are no containment isolation valves between the PCCS condensers and the containment, they system is always in “ready standby” mode.

BWRX-300 is being licensed in the US NRC using so-called licensing topical reports (LTRs) representing the major deviations relative to the already certified ESBWR. The first LTR that forms the basis for the major simplifications of the design was submitted in December 2019 and approved by the NRC in December 2020. Three more LTRs were submitted in 2020 which are expected to be reviewed and accepted in 2021. Pre-licensing is also underway in Canada where eight submittals of the 19 VDR focus areas were made in early 2020. GEH is simultaneously in the first and second phase of the three-phase VDR process. Pre-application activities are also being prepared in the UK.

The current deployment schedule aims for a commercial deployment date between 2027 and 2028.

Emergency Planning Zone

As the detailed design is not yet finished the source term estimation and emergency planning zones sizing is still to be calculated.

The initiating events GEH is considering are similar to those used in the ESBWR certification process[35]. Use of a mechanistic source term for the BWRX-300 is planned to follow the general path of most SMRs and is being supported in the US by the Nuclear Energy Institute (NEI)[36] (visualized in Figure 17). A bounding worse case source term during design is considered and, to the fullest extent possible, GEH designs to generic sites so that the same design may be used in various licensing regimes. However, site specific confirmations are required for individual projects. The LTR on source term is being prepared for submissions to the NRC in US and CNSC in Canada.

[35] GEH Nuclear Energy, “ESBWR Design Control Document”, Rev. 10, Ch. 6, 15, 2014.
https://www.nrc.gov/reactors/new-reactors/design-cert/esbwr.html
[36] Nuclear Energy Institute, “Small modular reactor source terms”, NEI, December 27, 2012.
https://www.nrc.gov/docs/ML1336/ML13364A345.pdf

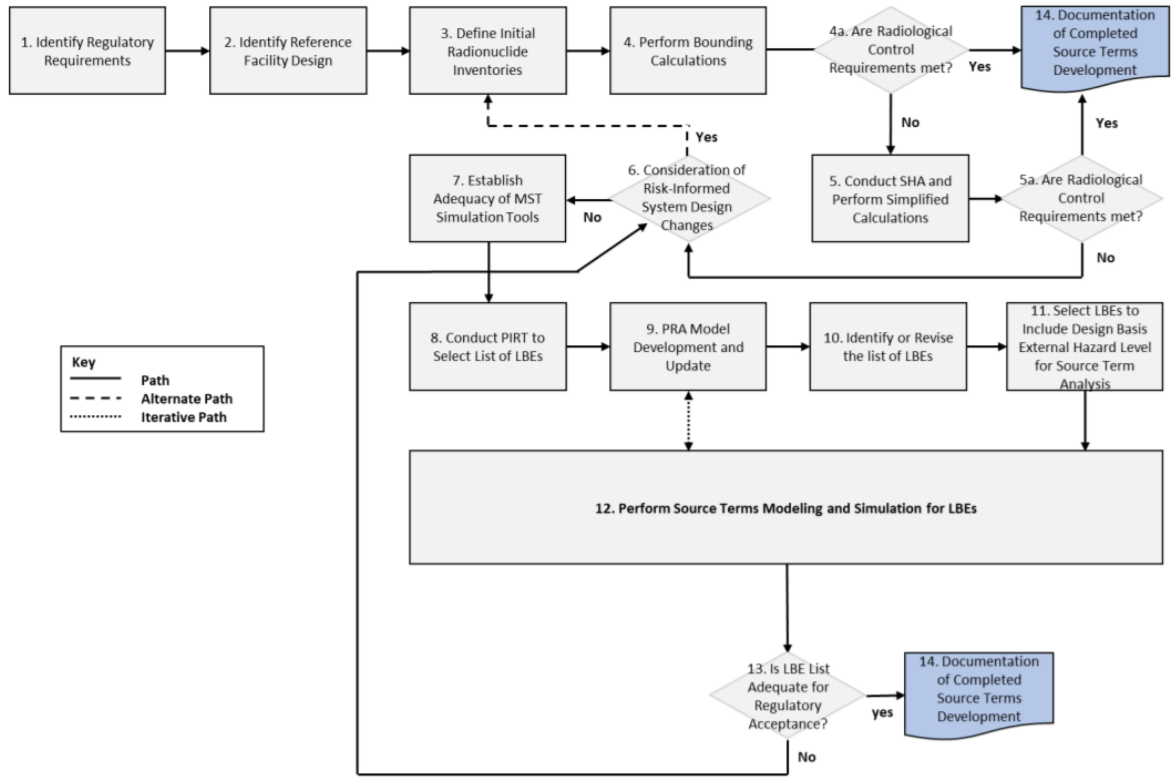

Figure 17: Technology-inclusive source terms determination methodology[37].

It is envisaged that the EPZ sizes will be calculated using dose-based and consequence-oriented methods and, according to GEH, the TVA Clinch River Early Site Permit process is a representative estimate.

The NRC is currently updating its regulations to provide technology-inclusive, risk-informed, and performance-based criteria for the assessment of population-related issues in siting of advanced reactors[38]. This pertains to both, source term assessment as well as emergency preparedness and planning including EPZ sizing. Depending on the timing of relevant LTR submissions, GEH may be able to use the updated guidelines.

Integral Molten Salt Reactor (Terrestrial Energy Inc)

The Integral Molten Salt Reactor (IMSR®) is a pool-type molten salt SMR designed by the Canadian company Terrestrial Energy Inc (TEI). The IMSR® is a graphite-moderated thermal-

[37] US NRC, "Advanced Reactor Stakeholder Public Meeting", Slide 59, August 2020. https://adamswebsearch2.nrc.gov/webSearch2/main.jsp?AccessionNumber=ML20232D114

[38] NRC, "Small Modular Reactor and Non-Light Water Reactor Technical and Policy Issues". https://www.nrc.gov/reactors/new-reactors/smr.html

spectrum burner-type reactor using molten fluorides as fuel and coolant. Reactor power is 195 MWe or 440 MWth.

The IMSR® power plant features a completely sealed reactor vessel Core-unit. All primary reactor components, including pumps and heat exchangers are integrated into the sealed and replaceable Core-unit, with the reactor vessel and its closure head forming the primary boundary (Figure 18). There are no external primary system piping loops, no external primary system pumps, and no pressurizer of any kind. The nuclear fuel and coolant circulate entirely within, never exiting, the reactor vessel. The high temperature heat is transferred from the Core-unit through a sequence of isolated non-radioactive molten salt loops to the final process application. Penetrations at the top of the Core-unit provide makeup fuel additions and secondary coolant circulation. The Core-unit operating lifetime is 7 years. After this period, a new Core-Unit replaces the spent unit. This approach eliminates any need to open the Core-unit for graphite replacement, maintenance and repairs. The first IMSR® power plant will be a single reactor facility, but the system is easily scalable to include multiple reactors on the same site each with its own Core-unit.

The IMSR® power plant can be used to generate electricity through a conventional Rankine cycle. Depending on the customer needs, high-temperature heat can also be stored and/or used for industrial processes such as hydrogen, industrial steam, synthetic fuels and desalinated water production.

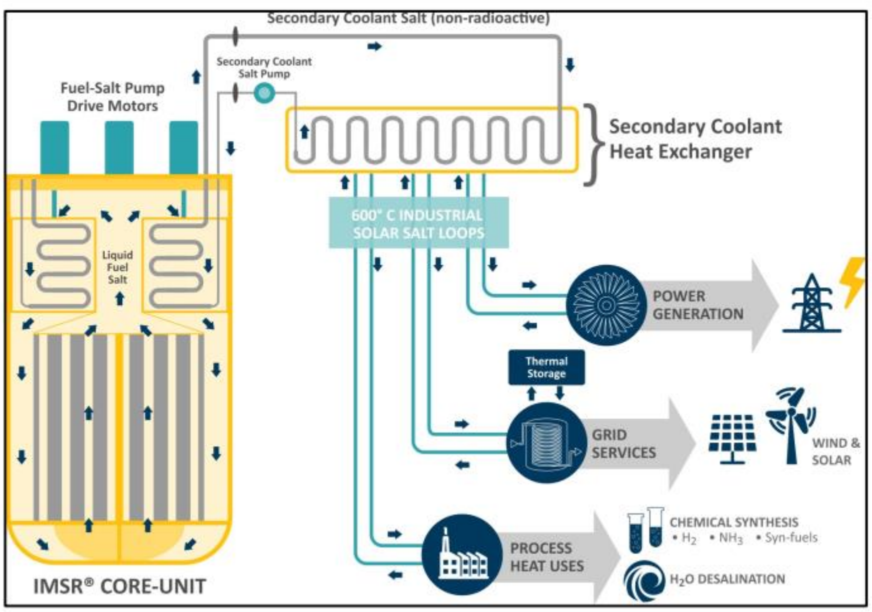

Figure 18: A functional schematic of the IMSR® power plant.

An IMSR® power plant has commonly two operating silos enabling the switch to a new Core-unit every seven years (Figure 19). Each silo consists of a containment that accommodates a guard vessel designed to support the Core-unit and add a layer of containment. Together with the containment the guard vessel acts as a barrier for any leaked fuel salt or radioactive material to escape to external spaces within the plant. The guard vessel is part of the containment and designed to last for the whole operating time of the plant. The focus of TEI is to build a single unit plant first but multi-unit facility is possible should a customer need that.

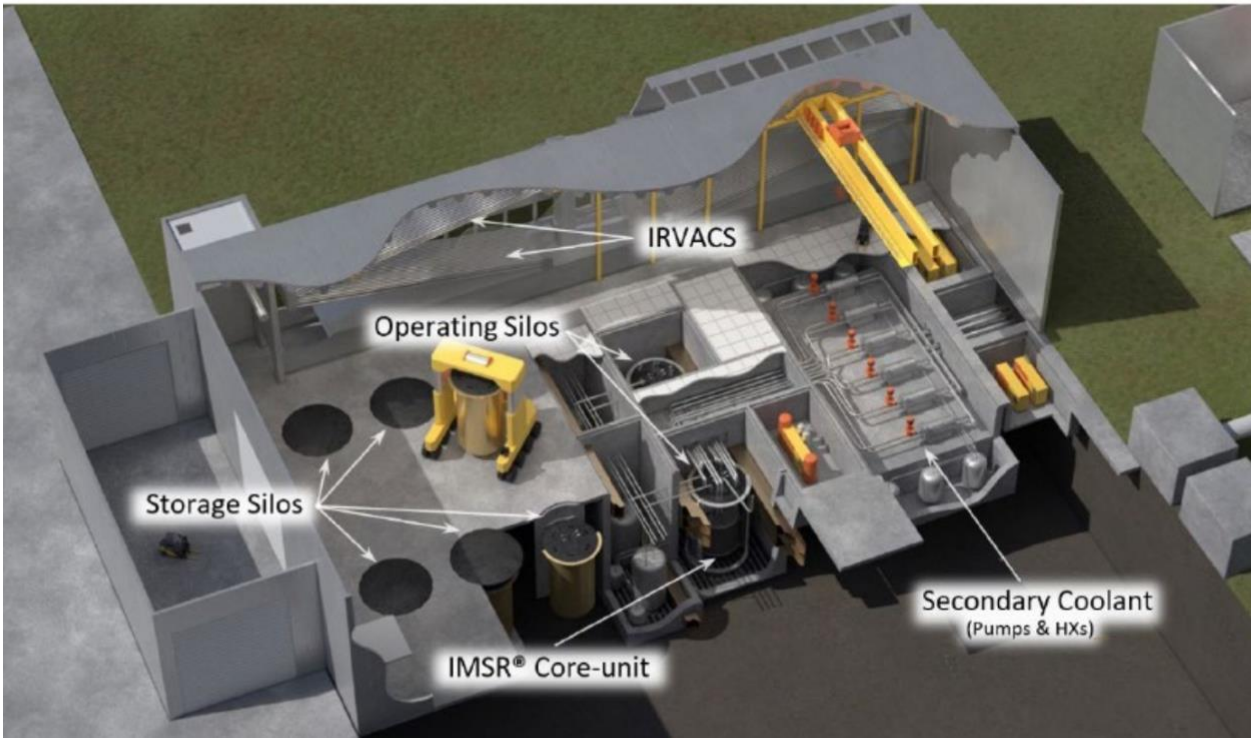

Figure 19:A computer render of the IMSR® power plant.

A “generic design site envelope,” is used to develop the IMSR® site design and encompasses generic site parameters used in Canada, the U.S., and European countries relevant to nuclear plant siting. The structures surrounding the reactor such as auxiliary systems, buildings, fences, and the RAB until the site boundary are shown in Figure 20.

The safety of IMSR® is based on its intrinsic design features. There is no intention to formulate a specific list of passive safety systems rather the “control, cool, contain” targets are achieved through several passive features and capabilities that, in aggregate, provide for passive plant control and response.

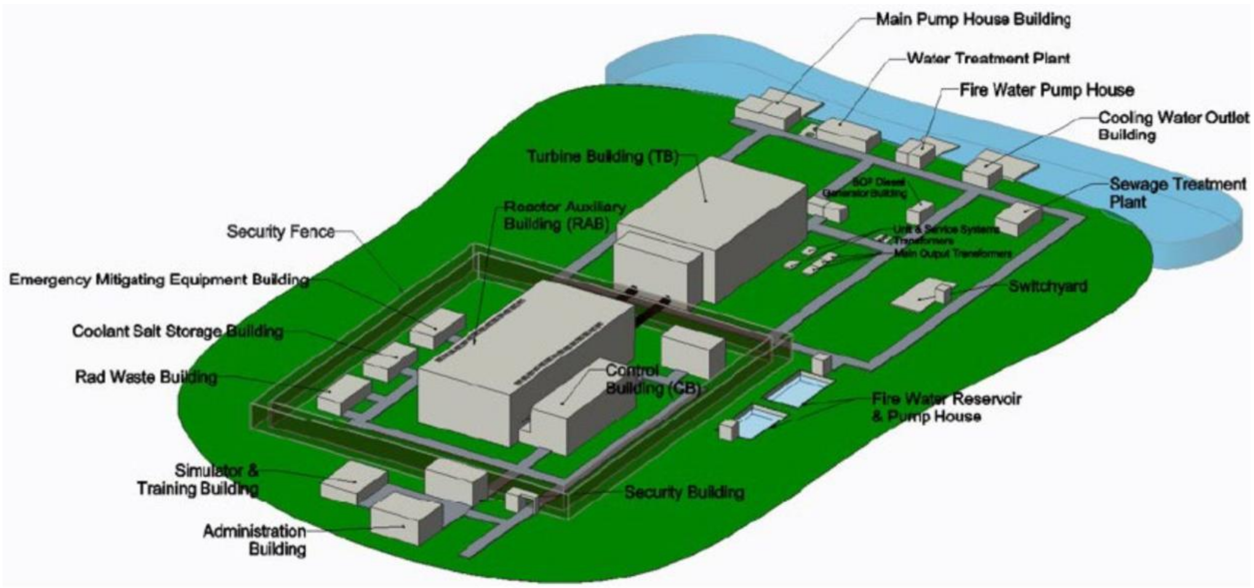

Figure 20: The IMSR® plant site layout.

The IMSR® uses commercially available low-enriched uranium (<5%) as the fuel. Uranium tetrafluoride (UF4) fuel is dissolved in a low-cost salt eutectic mixture to form an integrated fuel-coolant mixture. As no solid fuel elements are used, heat is generated directly within the fuel-coolant fluid. Therefore, the heat transfer takes place majorly via convection resulting in relatively small thermal gradients. The reactor operates at a near atmospheric pressure, which means that no strong high-pressure equipment and structures are needed. Heat removal by natural convection is possible in various regimes of operation. High heat capacity increases the thermal inertia of the system. The fluoride salts used in IMSR® form strong chemical bonds with the most radioactive fission products retaining them in the fluid. It is just the noble gases xenon and krypton that have low solubility in the salt and minor presence in the graphite. These common neutron poisons will be constantly vented out improving thereby also the load following capability of the reactor system. These salts have a very high boiling point (>1400 C) and low vapor pressure reducing the risk of releasing the radioactive materials from the molten salt.

The IMSR® reactor does not require control rods for the criticality control, which is controlled through negative temperature feedback effects, similar to most reactors. This means that when the temperature increases, the fuel-coolant density decreases, and nuclear absorption becomes stronger resulting in decrease of reactor power. In a potential emergency situation, the residual decay heat is passively removed from the Core-unit using the Internal Reactor Vessel Auxiliary Cooling System (IRVACS). IRVACS is an inert-gas circulation system capable of transferring maximum decay heat levels to the atmosphere.

The licensing efforts are primarily focused in Canada where the TEI is engaged in the CNSC VDR process. VDR Phase 1 was completed in 2017 and the Phase 2 is now progressing and is planned for completion in 2021. At the same time, TEI's US affiliate TEUSA is engaging in pre-licensing activities with the US NRC. The engagement activities are guided by a formal Regulatory Engagement Plan (REP) that addresses the key strategic elements of the licensing process necessary to support the submittal of the license application for the IMSR®.

In August 2019, the NRC and the CNSC signed a Memorandum of Cooperation (MoC) to establish a collaborative joint review process for the advanced reactor technologies. The IMSR® licensing strategy is well aligned with this MoC and considering the simultaneous pre-licensing activities in Canada and US, TEI believes it is well placed to pursue a "two-country" license.

Emergency Planning Zone

The postulated initiating events (PIEs) used in the design and safety analysis of the IMSR® are identified based on a systematic approach supplemented by Molten-Salt Reactor Experiment (MSRE) experience and engineering judgement. Using multi-level top-down and bottom-up approaches, a comprehensive list of PIEs was developed. This list is compared against other similar types of reactors for further confidence.

The IMSR® safety analysis will be carried out using combined probabilistic-deterministic safety and risk assessment approaches. Based on the preliminary results, the most severe DBA is the off-gas line break that may lead to release of radioactivity out of the containment. This event falls within the DBA frequency range (as defined by the Canadian regulatory framework) of between 10^{-5} and 10^{-2} per reactor year, with the exact value to be determined in the detailed probabilistic safety analysis as the design evolves.

As common to most SMRs, TEI is planning to use a mechanistic approach for the IMSR® source term estimation. As most fission products and actinides are retained within the salt, the source term is mostly composed of the noble gases accumulating in the upper plenum. The estimated source term is then used to calculate the doses to an individual or to the population should the off-gas break DBA happen. This was done using ADDAM (Atmospheric Dispersion and Dose Analysis Method) code and development of a fully integrated system code SPECTRA (Sophisticated Plant Evaluation Code for Thermal-hydraulics Response Assessment) is ongoing for future design and safety analysis including off-site consequences.

In both licensing regimes (CNSC and US NRC), the EPZ size will eventually be determined by the licensee based on the safety analysis and the regulatory requirements of each country. The final EPZ takes into account considerations from other relevant parties (radiation protection, health, environment agencies as well as local communities). In principle, the doses resulting from the bounding accident scenarios have to remain below the regulatory limits enforced by the CNSC and/or NRC.

The IMSR® design target is that the EAB remains within the site boundary. This is mostly due to operational requirements as opposed to safety requirements. Preliminary conservative safety analysis of the IMSR® conceptual design shows that the most bounding DBA results in doses of an order of magnitude lower than the regulatory limit at 200 m. The dose limit for the anticipated operational occurrences (AOOs) is 0.5 mSv and for the DBAs is 20 mSv.

TEI believes the IMSR® design target of no off-site EPZ is feasible and can be technically proven. TEI has projected that most of the PIEs will have minimal or no consequences because of the fundamental nature of the design and passive/inherent safety features of the IMSR®. This would result in a significantly reduced EPZ in either country whilst still recognizing and meeting the jurisdictional requirements specific to each country.

MMR (Ultra Safe Nuclear Company)

The Micro Modular Reactor (MMR) a helium-cooled high-temperature thermal spectrum reactor developed by the Ultra Safe Nuclear Company (USNC). Two versions, 15 MWth (5MWe) and 30 MWth (10 MWe) with operating fuel lifetime of 20 years and 10 years, respectively. The MMR facility consists of a nuclear plant that provides the heat and an adjacent plant where the heat is converted into electricity, used for some industrial process, or stored (Figure 21).

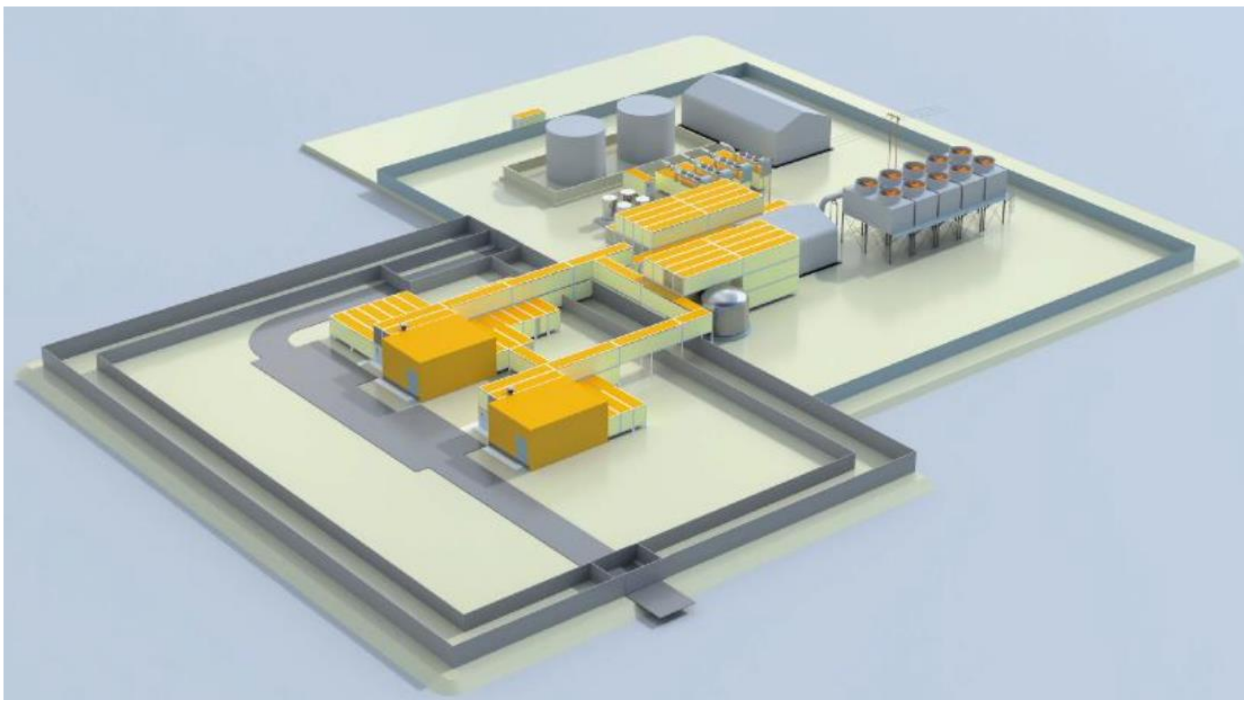

Figure 21: USNC MMR nuclear plant (left) and adjacent plant (right).

The nuclear heat supply system consists of a reactor, hot gas duct, hot gas fan and an intermediate heat exchanger forming a closed loop filled with helium (Figure 22 and Figure 23). Heat from the reactor is transferred to the heat exchanger via hot gas flow driven by an electric fan.

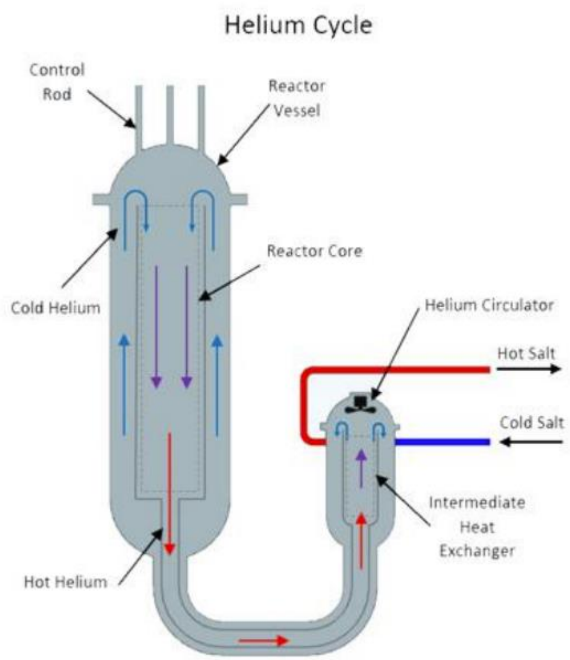

Figure 22: MMR reactor helium flow path.

Figure 23: Rendering of the MMR reactor model.

In the heat exchanger, the heat is given to an intermediate loop operating with up to 565 °C molten salt. This loop also contains thermal storage equipment. From the molten salt loop, the heat is transferred to the adjacent plant for selected process applications. An example of electricity generation set up is shown in Figure 24. Power conversion with up to 33% efficiency.

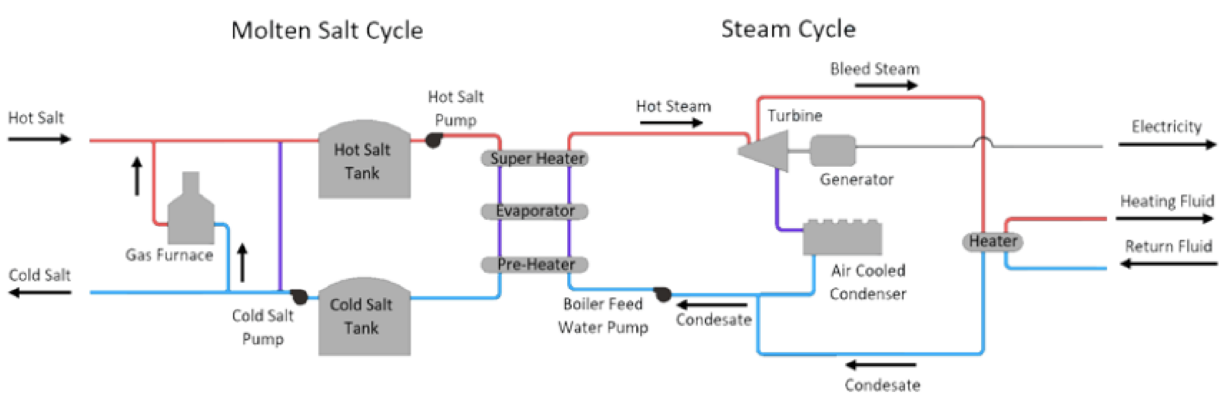

Figure 24: A simplified process diagram of electricity generation in the adjacent plant.

Reactor building (or citadel building as USNC names it) is about 10 meters high and located below grade (Figure 25). A facility consisting of two MMR reactors together with associated gas cooling systems and a single steam turbine fit into a 100 m x 200 m area.

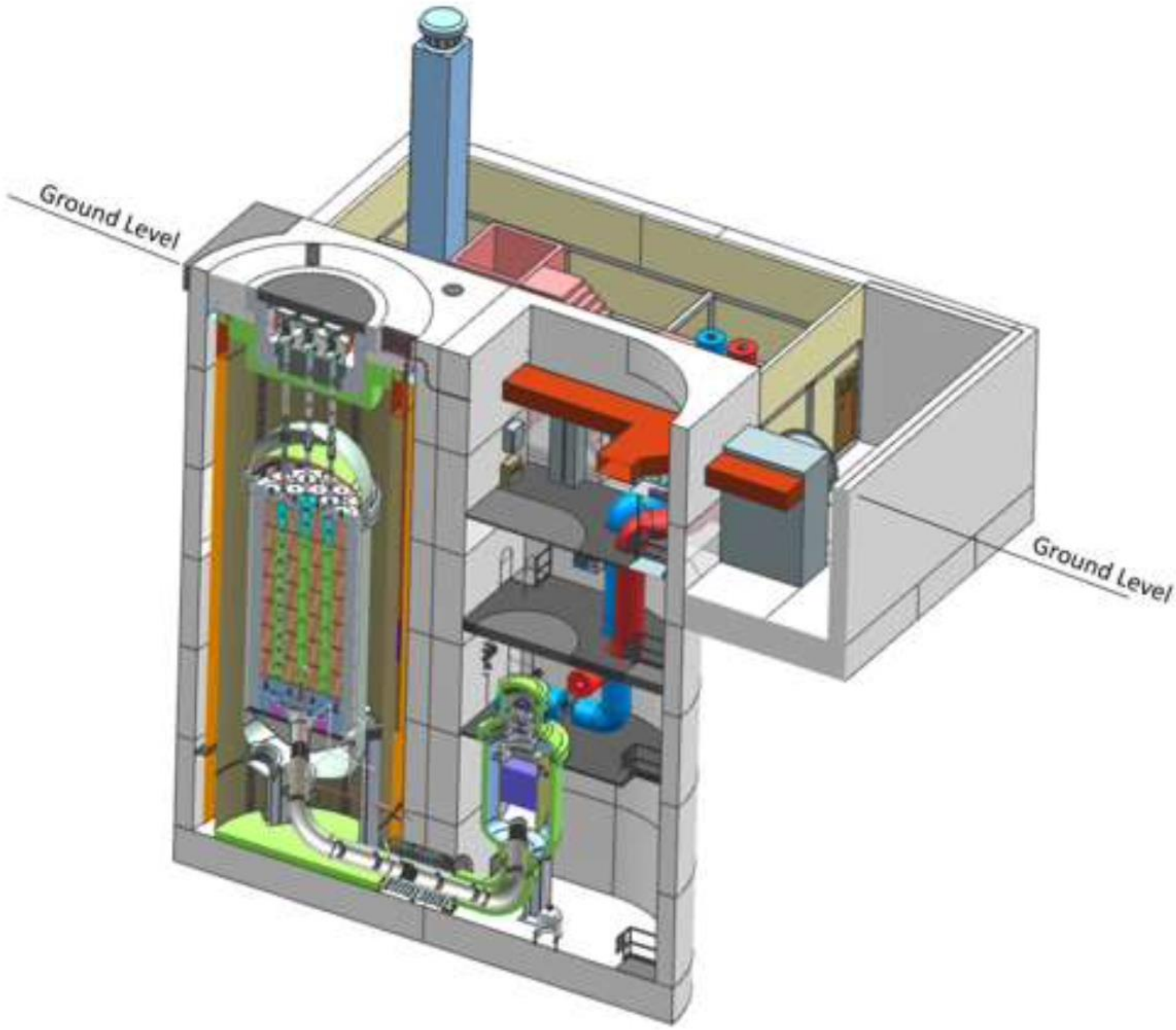

Figure 25: MMR reactor building (citadel) located below the ground level.

The MMR is fabricated in harmonized modules (size of a standard shipping container) that can be assembled, commissioned and tested off-site before transporting and installing the system at the customer's site. The modules would include necessary piping, cabling, lighting etc. in order to minimize the on-site construction work. The same applies to pre-cast concrete structures to the extent practical.

The MMR reactor uses 0.5 mm diameter TRi-structural ISOtropic (TRISO) particles encased in a silicon carbide matrix to form what is called Fully Ceramic Micro-encapsulated (FCM) fuel, size of about few cm as the pellets in conventional LWRs. Altogether, the reactor core consists of hexagonal graphite blocks penetrated by full length channels for the FCM fuel pellets, helium flow and control rods. The graphite blocks function as structural support, neutron moderator and reflector. See Figure 26 for fuel-to-core configuration.

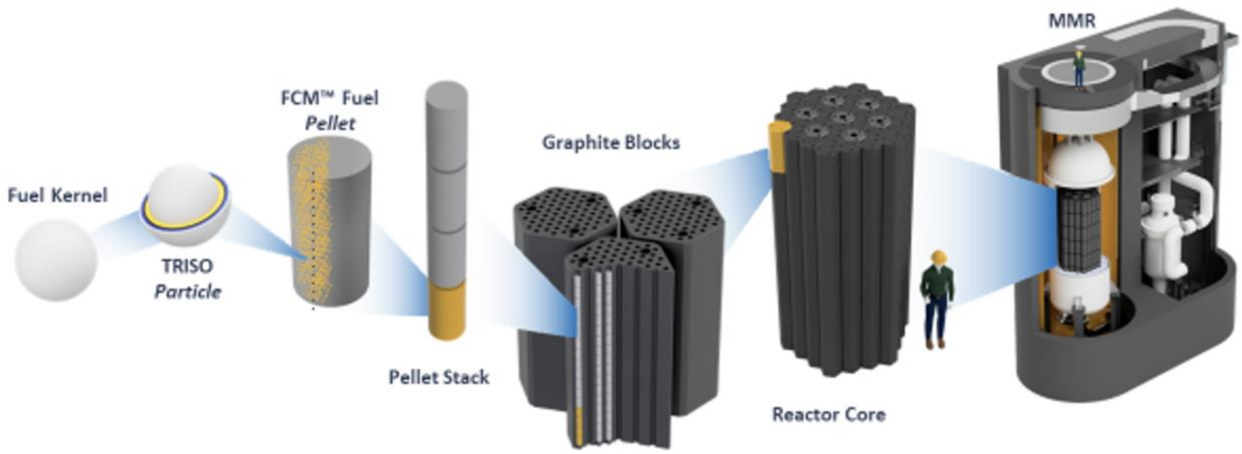

Figure 26: Conceptual arrangement of the MMR core consisting of TRISO particles, FCM fuel and graphite blocks.

Fundamental basis for ceramic coated particle fuel technology was developed in the 1960s[39]. An important advantage of the TRISO fuel particles is that having multiple layers of ceramic material around the nuclear fuel kernel provides good retention of fission products. This type of fuel has been deployed in early high temperature gas cooled reactors. In 2002, US DOE's Advanced Gas Reactor (AGR) Fuel Development Program started to develop advanced fabrication and characterization methods and provide irradiation and safety performance data required for licensing the future advanced gas reactors[40]. Nearly 300,000 particles were exposed to fast neutron irradiation and temperatures up to 1600 °C. The results indicated record burnups (several times what is achieved in current LWRs) and no particle failures. This indicates that the probability of radioactive material leakage is very low in any reactor situation i.e., normal operation or accident. However, the fuel qualification work in the US is ongoing involving state support, several national laboratories and private companies.

The safety case of MMR design is built upon the performance of the FCM fuel, ~100 times smaller core and very low power density (1-3 W/cm^3 vs 20-40 W/cm^3) compared to conventional LLWRs. The FCM contains multiple barriers for fission products (fuel kernel, TRISO coatings, FCM SiC matrix). While retaining the fission products the ceramic fuel still conducts heat. Combined with low power density and applied engineering practices, core melting can be excluded under any circumstances. Two additional barriers are provided by the helium pressure boundary and the concrete reactor building. Reactivity control is achieved through a negative temperature-reactivity feedback coefficient meaning that when the reactor temperature exceeds certain pre-determined

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[39] Demkowicz et al., "Coated particle fuel: Historical perspectives and current progress," J. Nuc. Mat, Vol 515, 434-450, 2019.

[40] Demkowicz et al., "Results of the AGR-2 TRISO fuel performance demonstration irradiation experiment in the Advanced Test Reactor,“ Ann. Nucl. En., Vol. 150, 2021.

limit (well below melting temperature of any reactor material) the nuclear reactions are stopped. Additionally, gravity driven reactor control rods are used. Decay heat management is based on the initial heat redistribution in the surrounding structures (graphite, RPV and reactor building) (note that the reactor building outer walls are assumed adiabatic) and subsequent removal of the heat to the atmosphere using a reactor cavity cooling system (RCCS). The detailed dynamics of heat distribution and PCCS performance will be investigated in the future.

USNC engineering approach to calculate the risk as the product of probabilities and consequences aims to keep the probabilities of adverse events low and eliminate or minimize the serious consequences. Preliminary deterministic safety analysis indicates that the temperature of the fuel, RPV and reactor building remains with sufficient margin below the design limits for normal operational transients and DBAs.

USNC has also analyzed certain BDBE events, notably the chimney break (equivalent to a hot duct cross-section size break) and found that the key temperatures remain below safety limits provided that a coolable core geometry is maintained. Should extensive graphite corrosion lead to failure of the core blocks it is still expected that the FCM fuel retains its integrity given the low power density, demonstrated temperature resistance up to 1800 °C and very low possibility of re-criticality in the collapsed geometry. Detailed analysis of such a situation is yet to be carried out.

USNC is currently in the CNSC VDR process in Canada. Phase 1 has been completed with the regulator confirming that the USNC interprets the regulation appropriately, but additional information is required to confirm the:

- adequacy of the MMR R&D activities
- applicability of the operating experience from earlier gas cooled reactors to the MMR
- consistency between the safety functions and the safety classification for the structures, systems and components related to civil structures
- adequacy of shutdown means, shutdown margins and the guaranteed shutdown state
- adequacy of the proposed PRA methodology

to substantiate MMR safety claims and the fuel qualification program, including the role of a first-of-a-kind reactor.

At the same time, USNC is the vendor partner in a joint venture Global First Power together with the Ontario Power Generation as the utility partner. The goal is to license and build the MMR at the Canadian Nuclear Laboratories' Chalk River site in Ontario.

USNC is also proposing to build MMRs at several sites in the US. The US projects can benefit from the Canadian project, as the design verification and licensing work with the CNSC will be shared with the US NRC as facilitated through a cooperation agreement between regulators.

Emergency Planning Zone

USNC is planning to carry out deterministic and probabilistic hazard and risk analysis according to the requirements of the jurisdiction the reactor is being licensed. The MMR source term will be based on the radioactive release estimates related to fuel degradation. These estimates will be based on experimental data on fuel fracture fractions obtained in the tests with TRISO as well as FCM fuel elements. The resulting bounding values will be used to estimate the dose levels outside the reactor.

Based on the calculations performed to date, USNC can claim that the size of the exclusion zone falls within the fenced site perimeter. It is estimated to be approximately 30 meters from the reactor building where the dose acceptance criteria are met for all AOOs and DBAs.

Even though the final sizing will be determined during actual site selection and requires more detailed safety analysis, the design goal of the MMR is to have no need for emergency evacuation or sheltering of off-site individuals during PIEs. At the operator's discretion, nonessential plant personnel will be relocated from the site during abnormal events to avoid worker exposure.

UK SMR (A consortium led by Rolls Royce)

UK SMR is a project aiming to develop a market-competitive, factory-made reactor utilizing the vast experience of pressurized water technology. The UK SMR is a three-loop PWR producing approximately 450 MWe of power using industry standard UO2 fuel enriched up to 4.95%. Core design (UO2 pellets, zirconium-alloy cladding, assembly and core configuration) is based on conventional PWR technology. The coolant is circulated by three centrifugal seal-less pumps between the core and three vertical u-tube steam generators (Figure 27).

The power plant's nuclear island, turbine island and cooling water pumping facility are all protected by a robust hazard shield (Figure 28). Support buildings and those containing auxiliary services are situated within a berm that sweeps around the site and provides further protection from external hazards e.g., tsunami or aircraft impact.

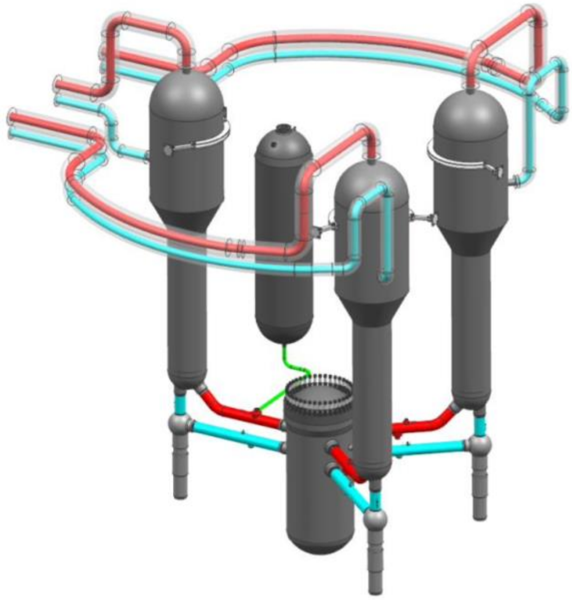

Figure 27: 3-loop UK SMR coolant system with a pressurizer connected to one of the loops (green line).

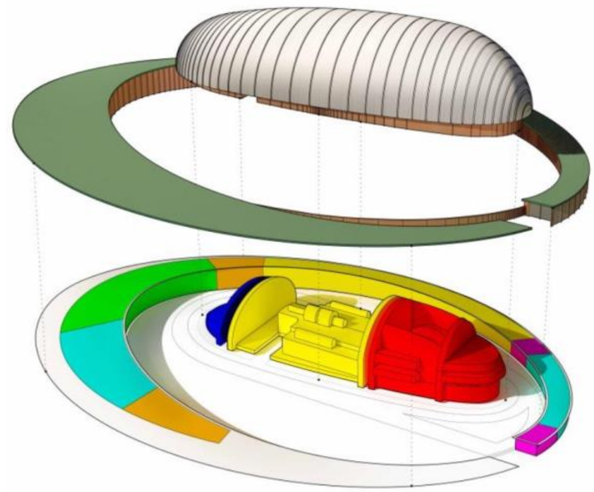

Figure 28: UK SMR expanded site view. Nuclear island (red), turbine island (yellow) and pumping facility (blue).

The reactor shall be fully modularized, transportable by road, rail or sea and have a target construction time of four years (two years for the preparation and civil groundworks, two years for the on-site reactor assembly). RPV is 11.3 m in height and no more than 4.5 meters in diameter (constrained by UK road transport height limit of 4.95 m). As with most water based SMRs, the primary intention is grid-scale stable electricity production, but the design could be configured to provide district heating, process steam, water desalination or hydrogen.

The UK SMR duty reactivity control is provided by the control rods and the negative moderator temperature coefficient common in PWRs. No soluble boron is used for this purpose leading to potential design simplifications.

The design includes multiple diverse and redundant independent active and passive safety systems. Heat removal from the core takes place via the steam generators, the Passive Decay Heat Removal (PDHR) system and the Emergency Core Cooling System (ECCS). All design basis LOCAs are protectable by ECCS, with diverse protection additionally available from the Small Leak Injection System (SLIS) for smaller leaks. Control and scram rods and emergency boron injection provide two diverse and highly reliable means of reactor shutdown. Three Safety Relief Valves (SRVs) are used to protect against overpressure hazards, each fully capable of providing relief. Robust containment, including a core catcher, is provided to mitigate the release of fission products to the environment in the unlikely event of core damage.

Preliminary Probabilistic Safety Analysis (PSA) shows that the Core Damage Frequency (CDF) from all plant hazards is below 1 x 10^{-7} per year of operation (more than 10 times less than required by the ONR). No single event causing disproportionate risk of core damage has been identified. The design philosophy prioritizes passive systems overactive ones meaning that the external electrical supply and human actions do not contribute significantly to the UK SMR risks. Automatic initiation and dimensioning of the safety systems shall ensure that no active intervention is needed within the first 72 hours of any DBA/DEC.

The UK SMR project is a collaborative effort of British nuclear design, engineering and infrastructure companies - a consortium led by Rolls Royce. Partners of the consortium include Assystem, Atkins, Jacobs, National Nuclear Laboratory, Rolls Royce, Laing O'Rourke, TWI, Nuclear AMRC, BAM Nuttall.

The design is currently at a mature concept state and the consortium is finalizing the basis to prepare for the entry into the Generic Design Assessment (GDA) Phase 1 in Spring 2021. Once all phases of GDA are approved, it only remains to address the site-specific aspects by the utility building the SMR (it is estimated that about 85% is standardized and 15% is site specific). A successful GDA process would result in a Design Acceptance Confirmation (DAC) that could be referred to in licensing, building and operating the plant by any customer. Additionally, the UK SMR will be assessed in the IAEA Generic Reactor Safety (GSR) review. The goal is to deploy first of a kind UK SMR in the UK by 2030.

Emergency Planning Zone

The UK regulatory framework requires that safety functions to prevent or mitigate radiological hazards are categorized based on the consequences and likelihood of failure of the safety function (Cat A - most important, Cat C - least important). Similar is required for all structures, systems and components that are responsible for fulfilling the safety functions (Class 1 - highest, Class 4 - lowest).

The safety case for the SMR is being developed using a systematic approach based on qualified methods to ensure that all types of hazards (LOCA, non-LOCA, internal and external hazards) are identified and that sufficient safety measures are in place to ensure that the people and the environment are protected. The safety case addresses in detail all potential hazards which are then screened on frequency and consequence to formulate a design basis set of Postulated Initiating Events (PIEs) that will be analyzed deterministically. Hazards of very low risk i.e., aggregate of likelihood and consequence, will not be considered further.

Regarding source term, a Radioactive Source Term Policy document has been produced. The principles described in that policy are used to ensure that normal operation and accident doses will remain sufficiently below the required criteria. As of now, no dose assessment has been done but an initial best estimate source term has been estimated based on operational experience from existing plants. A design-specific source term is being calculated as it is one of the key elements in the safety analysis and emergency planning.

No work has been done to date on defining the EPZ for the UK SMR. EPZs will be determined by the ONR judgement which combines technical assessment of the plant (e.g., source term) and its operating procedures, site specific factors, together with other factors considered by the ONR Emergency Preparedness and Response Team, including IAEA guidance.

6. Clinch River Early Site Permit

In May 2016, Tennessee Valley Authority (TVA) submitted an early site permit (ESP) application. The proposed site, which hosted the former Clinch River Breeder Reactor Project, is located on a tract of land adjacent to the Clinch River arm of the Watts Bar Reservoir. TVA requested an ESP with a permit duration of 20 years from the date of issuance.

TVA stated that it is currently evaluating four light-water-cooled SMR technologies for deployment at the CRN site. Because a reactor technology has not been selected for deployment at the CRN Site, the plant-site interface is defined through a plant parameter envelope (PPE). This is meant to provide sufficient design detail to describe each possible technology in accordance with both the NRC safety and environmental review of the ESP application.

The four conceptual SMR designs that were used to create a “surrogate plant” as defined in NEI 10-01, Industry Guideline for Developing a Plant Parameter Envelope in Support of an Early Site Permit. The four conceptual SMR designs considered were:

- BWX Technologies, Inc. (BWXT) mPower™ (Generation mPower LLC design)
- NuScale (NuScale Power, LLC, design)
- SMR-160 (Holtec SMR, LLC, design)
- Westinghouse SMR (Westinghouse Electric Company, LLC, design)

All four designs are described as passively safe with minimal or no reliance on offsite power, offsite water or operation action for safety. The fact that SMRs are inherently safer due to the increased safety margin, smaller source term, reduced magnitude and probability of potential accident sequences, and the slower accident progression, moreover, various DBAs are eliminated by design, and BDBAs are less likely to occur.

The applicant provided, through its PPE, sufficient design information to allow it to perform the analysis required to determine exclusion area boundary (EAB) and low-population zone (LPZ) radii of the site. TVA selected the DBAs to determine the consequences in accordance to the four LWR technologies being considered.

As part of the application, TVA submitted two distinct major features emergency plans: ESPA Part 5A (site boundary EPZ) and ESPA Part 5B (2-mile EPZ). The application did not include off site Radiological Emergency Preparedness (REP) plans in support of the CRN Site, and stated that ingestion exposure pathway EPZ for the CRN Site will be described in the future in a combined license application (COLA).

The emergency plan (ESP Plan 5A or 5B) ultimately selected for the site in a future COL or CP application would be based upon the selected SMR design’s ability to meet the criteria in the applicable plan, including the PEP EPZ size. An appropriate PEP EPZ size would be established in a COLA or a construction permit (CP), should it be issued.

Source term determination

TVA's approach to determine the EPZ size starts by getting the exemptions requests for EPZ. In order to meet these exemptions, they started by selecting a surrogate design meaning that the parameters based on which the EPZ is evaluated are not reactor technology specific.

The technology-agnostic 4-day total atmospheric release source term was defined according to following:

1. TVA created a composite source term based on vendor information on accident source terms from a spectrum of accidents and a set of SMR vendors.
2. The composite source term is based on three reactors, one is a LLWR and other two SMRs designs. Afterwards, TVA took the largest magnitude of release of a period of 4 days. To account for design uncertainty and the current analysis maturity for all the SMRs, TVA increased the isotopic releases by a discretionary margin of 25 percent[41].
3. TVA used the obtained source term as input to an analysis. This included adjustments to the isotopic activity values for use as an input to the MELCOR Accident Consequence Code System computer code. These adjustments increased the margin to more than 25 percent.

After determining the source term release the dose associated to the release should not exceed the Environmental Protection Agency (EPA) Protective Action Guides (PAG), the verification steps are summarized as the following:

1. Select appropriate accident scenarios.
2. Determine source terms for selected accident scenarios.
3. Calculate the dose consequences for selected accident scenarios at the PEP EPZ boundary.
4. Compare the dose consequences for selected accident scenarios with the EPA PAG.

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[41] ML17291A052 - Clinch River ESP Phase B SE, Section 13.3 - Conduct of Operations. (147 page(s), 7/20/2018)

EAB determination

By NRC's Regulatory Guide 4.7, an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 250 mSv total effective dose equivalent (TEDE).

The exclusion area boundary (EAB) is delineated by the boundaries of the CRN Site and is designated to be the CRN property line. However, an analytical EAB based on the shortest distance between the effluent release point and boundary of analytical EAB for each of 16 compass sectors is used conservatively as 335 meters, and is used for atmospheric dispersion (χ/Q) modeling.

This distance is established based on the minimum distance between the release point and the analytical EAB such that an individual located at any point on the EAB boundary would not receive a radiation dose in excess of 250 mSv total effective dose equivalent over any 2-hour period following a postulated fission product release. The various analytical EABs can be encompassed by an ellipse fixed completely within the CRN property boundary.

Because the radiological dose is directly proportional to the χ/Q value and the χ/Q value decreases as a function of distance from the release point to the boundary of EAB, the analytical EAB dose bounds the dose at the encompassed ellipse-shaped EAB and the actual EAB.

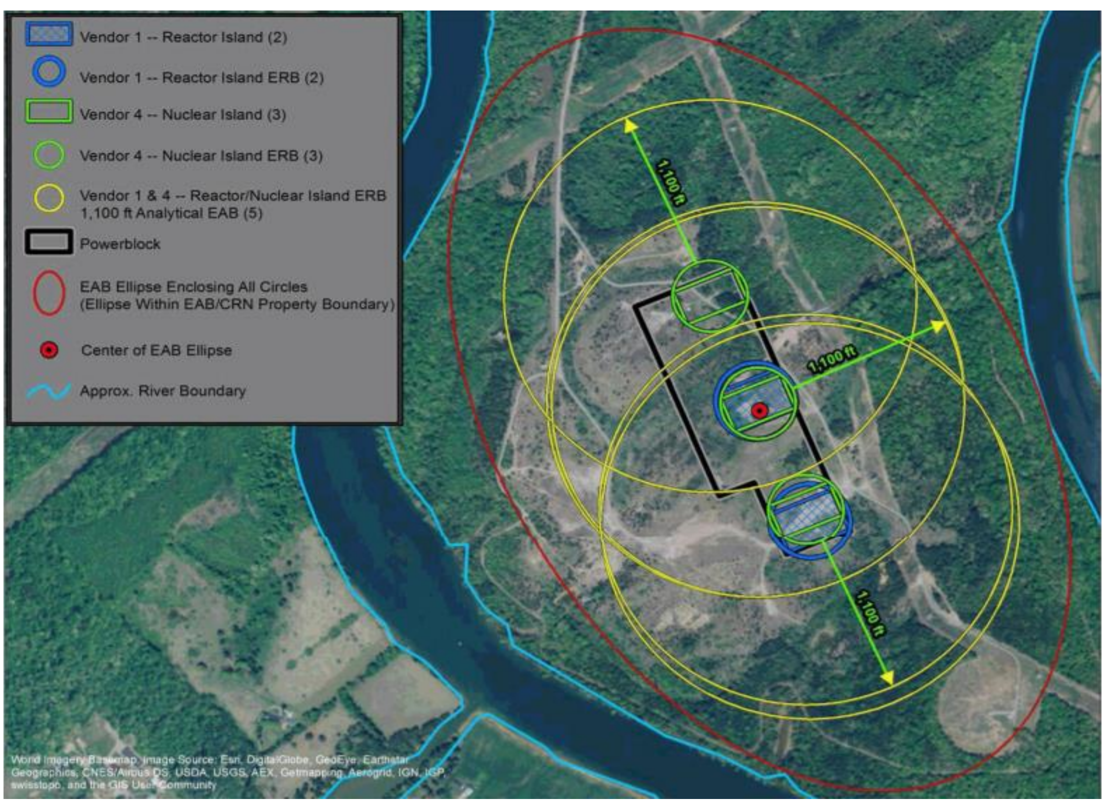

Figure 29: Effluent release zones with analytical EABs.

LPZ determination

To determine the low-population zone (LPZ) a description of the population distribution is needed. The description encompasses information about:

- the population in the site vicinity, including transient populations;
- the population in the exclusion area;
- whether appropriate protective measures could be taken on behalf of the populace in the specified low-population zone (LPZ) in the event of a serious accident;
- whether the nearest boundary of the closest population center having 25,000 or more residents is at least one and one-third times the distance from the reactor to the outer boundary of the LPZ;

Regulatory Guide 4.7 states that an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose over 250 mSv total effective dose equivalent (TEDE).

Population data should be estimated in relation to the time of initial plant approval, as noted above. Population projections should be considered over the lifetime of the facility. Further population projections should be made by decade for a 40-year period beyond the start of power plant operation.

Special population groups, such as those in hospitals, prisons, schools, or other facilities, that could have special needs during an emergency should be identified. In case of CRN Site there is one special facility identified as Kingston Academy, a 52-bed coed psychiatric residential treatment facility for children, is within the LPZ. There are no hospitals, prisons, or jails within the LPZ.

Conclusion

TVA notes that different reactor designs have different release pathways, and each pathway has different release rates and different radionuclide removal mechanisms. This is why TVA chose to use analyses from the design that resulted in the highest post-accident offsite doses in its assessment of radiological consequences at the CRN site.

TVA used the site characteristic short-term accident dispersion (χ/Q) factors at the exclusion area boundary (EAB) and the low-population zone (LPZ) boundary. TVA also presented the DBA dose assessment results at the proposed EAB and the LPZ outer boundary which show that the potential doses would remain under the threshold set forth in 10 CFR 50.34(a)(1) (TN249) and 10 CFR 52.17(a)(1) (TN251).

Because the reactor design technology is not selected and the orientations of plant structures on the site are not known, the detailed accident analyses and resulting post-accident doses for control room habitability and the Technical Support Center would be performed at the combined license application (COLA) stage. The ingestion pathway EPZ for the CRN Site would also be described in the application.

NRC staff concludes that TVA's proposed methodology for preparing an analysis to support the PEP EPZ size determination in a subsequent CRN site COLA or CP application is reasonable and consistent with NRC's considerations for SMR EPZ size determinations. The proposed methodology is acceptable for determining the appropriate size of the PEP EPZ for the CRN site.

At least one of the four SMR designs is expected to meet the dose criteria for the site boundary EPZ; all four are expected to meet the dose criteria for a 2-mile EPZ. This is TVA's stated expectation, but the NRC staff has not verified this claim as it is outside the scope of the ESP review.

In conclusion, the emergency planning zone distances also considered reasonable by NRC are as follows:

- Exclusion Area Boundary (EAB) - Clinch River Property Boundary
- Low Population Zone (LPZ) - 1 mi (1.61 km) from CNR Site center point

These values are given for a single unit but would be the same for each additional unit.

7. Summary and conclusions

Emergency preparedness and response planning is needed to protect the population and environment from hazards of an industrial activity. This is not a particularity of nuclear activities as chemical plants, for example, also require EPR measures to be in place.

The ICRP provides guidance on radiation protection in the form of reference levels for acceptable radiation doses. These are adopted by IAEA standards in the form of a comprehensive set of guidelines on EPR. Most regional and national nuclear regulators follow the IAEA and ICRP recommendations on radiation protection and EPR but incorporate local considerations in their own regulations.

The IAEA regulations on defining the EPZs and respective sizes are rather general, i.e., not prescriptive and found to be adequate for both large and small reactors. Countries where the national regulators have chosen to follow the IAEA guidelines, such as CNSC in Canada and ONR in the United Kingdom, have become the places where the vendors are seeking to license their new rector technologies. At the same time, the regulation in countries with typically large LLRW fleet tend to have evolved into a state which is adequate and efficient for large reactors but does not recognize the enhanced features of SMRs. A good example is US, where the current requirements for 10- and 50-mile EPZs are being revised towards a more performance-based, technology-inclusive, risk-informed, and consequence-oriented approach for EPZ and source term estimation. There is also a world-wide trend to harmonize regulations through collaboration (e.g., US NRC and CNSC joint review and licensing efforts).

The flexibility and technology-awareness of the regulation is important as it allows for appropriate EPZ sizing. This means that the SMRs with smaller power, advanced safety systems and other technological features do not potentially need an EPZ or require small EPZ. This makes it possible to locate them closer to consumers with less impact on the general public. Having production closer to consumption can lead to reduced heat and/or power losses as well as reduced needs for infrastructure development.

Among the reviewed five SMRs, all have provided evidence to a certain extent that their SMRs have a set of features that contribute to the minimization of EPZ, sometimes down to distances that reside within the site boundary.

The first demonstration of SMR siting and construction, according to the current knowledge, will take place in Canada where the Global First Power is working with Ontario Power Generation and USNC to deploy the MMR reactor project at Chalk River nuclear site. Environment assessment studies start in 2021 and the target construction start time is around 2024. The construction time is planned to be 1 year, and the low-power reactor will operate 20 years without refueling. Another relevant example to Estonia is the TVA Early-Site Permit process in the US which demonstrates how an SMR plant can be sited with EPZ at site boundary or have a 2-mile radius.

Based on the reviewed evidence (i.e., technology side), site bounded EPZ claims are realistic. Final answer will, however, depend on the local conditions.

Acknowledgements

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Glossary

ADDAM - Atmospheric Dispersion and Dose Analysis Method

AGR - Advanced Gas-cooled Reactor

ALARA - As Low As Reasonably Achievable

AOO - Anticipated Operational Occurrences

AMR - Advanced Modular Reactor

AST - Alternative Source Term

BDBA - Beyond Design Basis Accident

BWR - Boiling Water Reactor

BWRX-300 – Boiling Water Reactor X-300

CDE - Core Damage Event

CDST - Core Damage Source Term

CNSC - Canadian Nuclear Safety Commission

COLA - Combined License Application

CP - Construction Permit

CRN - Clinch River Nuclear site

DBA - Design Basis Accident

DSA - Deterministic Safety Assessment

CRP - Coordinated Research Project

DAC - Design Acceptance Confirmation (UK)

DCD - Design Certification Document

DG - Draft Regulatory Guide (US)

DOE - Department of Energy (US)

DEC - Design Extension Condition

DEPZ - Detailed Emergency Planning Zone

DHRS - Decay Heat Removal System

DID - Defense in Depth

EAB - Exclusion Area Boundary

EAL - Emergency Action Level

ECCS - Emergency Core Cooling System

EPA - Environmental Protection Agency (US)

EPD - Extended Planning Distance

EPR - Emergency Preparedness and Response

EPZ - Emergency Planning Zone

ESBWR - Economic Simplified Boiling Water Reactor

ESF - Engineered Safety Feature

EAEC - European Atomic Energy Community (EURATOM)

FCM - Fully Ceramic Micro-encapsulated

FEMA - Federal Emergency Management Agency

FNEP - Federal Nuclear Emergency Plan (Canada)

FSAR - Final Safety Assessment Report

GEH - General Electric Hitachi

GDA - Generic Design Assessment (UK)

GSR - General Safety Requirements (IAEA)

IAEA - The International Atomic Energy Agency

IC - Isolation Condenser

ICPD - Ingestion and Commodities Planning Distance (IAEA)

ICRP - The International Commission on Radiological Protection

IMSR - Integrated Molten Salt Reactor

IPCC - The Intergovernmental Panel on Climate Change

IRVACS - Internal Reactor Vessel Auxiliary Cooling System

LLWR - Large Light Water Reactor

LWR - Light Water Reactor

LOCA - Loss-Of-Coolant Accident

LPZ - Low Population Zone

LTR - Licensing Topical Report

MHA - Maximum Hypothetical Accident

MMR - Micro Modular Reactor

MST - Mechanistic Source Term

NEA - Nuclear Energy Agency

NEI – Nuclear Energy Institute

NPM - NuScale Power Module

NRC - Nuclear Regulatory Commission (US)

NSCA - Nuclear Safety and Control Act (Canada)

OIL - Operational Intervention Level

ONR - Office for Nuclear Regulation
ONT - Other Nuclear Technologies
OPZ - Outline Planning Zone
ORO - Offsite Response Organization
PAG - Protective Action Guides
PAZ - Precautionary Action Zone
PCV - Primary Containment Vessel
PCCS - Passive Containment Cooling System
PDHR - Passive Decay Heat Removal system
PEP - Plume Exposure Pathway
PIE - Postulated Initiating Event
PNERP - Provincial Nuclear Emergency Response Plan (Canada)
PWR - Pressurized Water Reactor
RAB - Reactor Area Boundary
REPPIR - Radiation (Emergency Preparedness and Public Information) Regulations (UK)
RG - Regulatory Guide (US)
PSA - Probabilistic Safety Assessment
RCCS - Reactor Cavity Cooling System
RPV - Reactor Pressure Vessel
SAR - Safety Analysis Report
SDA - Standard Design Approval
SG - Safety Guides (IAEA).
SLIS - Small Leak Injection System
SMR - Small Modular Reactor
SRV - Safety Relief Valve
SSC - Structures, Systems and Components
SSR - Specific Safety Requirements (IAEA)
STDBA – Source Term Design Basis Accident
TEDE - Total Effective Dose Equivalent
TEI - terrestrial Energy Inc.
TRISO - TRi-structural ISOtropic particle
TVA ESP - Tennessee Valley Authority Early Site Permit
UNSCEAR - The United Nations Scientific Committee on the Effects of Atomic Radiation

UPZ - Urgent Protective action planning Zone

USNC - Ultra Safe Nuclear Company

VDR - Vendor Design Review