EASI SMR

D1.8 EPZ calculation for a multi-unit SMR site

Oleksandr Sevbo (Energorisk), Jury Kyrylenko (SSTC NRS), Adam Kecek (UJV)

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Executive Summary

Deliverable D1.8 is the part of the EASI-SMR project in WP1 Transverse topics for LW SMR acceptability and licensing, the Task 1.4 Co-location of SMRs. It brings together results from a detailed independent process of a bibliographical review; practical experience of the contributors related to application of deterministic and probabilistic safety assessments in the emergency preparedness area; and comparison of the current methodologies/approaches used for the determination of size of emergency planning zone for nuclear sites with multiple small modular reactors.

The main objective is to explore the main issues on co-location of SMR modules at a nuclear site. The respective issues covered in this deliverable are related to determining emergency planning zone depending on number of SMR modules.

The deliverable brings together results from a detailed independent process of a bibliographical review; practical experience related to application of deterministic and probabilistic safety assessments in the emergency preparedness area; and comparison of the current methodologies/approaches used for the determination of size of emergency planning zone for nuclear sites with multiple small modular reactors.

The size of EPZ depends on site-specific factors such as source terms, meteorological conditions, topography, and the planned protective actions implemented during accidental radioactive releases at varying distances from the source. The small power output of individual SMRs may result in a lower source term (compared to large reactors) to be considered when defining the emergency planning zone. However, for multiple-modules sites, the total source term must be carefully assessed, as it may offset the expected advantages for emergency planning. Apart from this, the general requirements for defining emergency planning zones remain largely consistent with those for other types of nuclear facilities.

Available methods to establish the size of EPZ are discussed, along with lessons learned from ongoing international projects:

- Scaling;
- Deterministic approaches;
- Probabilistic /risk-informed approaches.

The deterministic approach remains conservative and comparatively straightforward to apply, whereas probabilistic methods offer a more detailed but complex alternative. Taking into account lack of detailed PSA Level 3 for the SMR designs, the deterministic approaches were selected for further calculations of EPZ under Task 1.4. A set of necessary input parameters for EPZ sizing is also outlined, providing a foundation for further assessment at subsequent stages of multi-unit/multi-module EPZ planning. Preliminary, in order to demonstrate the impact of the accident release magnitude on the size of EPZs, a simplified calculations were performed, with different locations of SMR modules at site

Obtained results demonstrates that for a single-module release, the distance at which dose criteria were exceeded is approximately 0.5 km, for 8-module release, the distance is almost 3 times extended. These results indicate that the number of modules may directly affect the EPZ size if simultaneous accidents at all modules are assumed. The calculations provide a useful first insight into the impact of the multi-module

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configuration on EPZ sizing, including a sensitivity analysis based on module placement relative to wind direction.

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Keywords

Emergency planning zones, co-location, radiological consequences, offsite doses, scaling, deterministic approach probabilistic approach, multi-module SMR

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Abbreviations and acronyms

Acronym	Description
AOO	Anticipated Operational Occurrence
BOC	Beginning of Campaign
CFD	Computational Fluid Dynamics
CRP	Coordinated Research Project
DBA	Design Basis Accident
DECs	Design Extension Conditions
DEC-B	Design Extension Conditions (Category B)
DSA	Deterministic Safety Assessment
EOC	End of Campaign
EPD	Emergency Planning Distance
EPR	Emergency Preparedness and Response
EPZ	Emergency Planning Zone
IAEA	International Atomic Energy Agency
ICPD	Ingestion and Commodities Planning Distance
iPWR	Integral Pressurised Water Reactor
IRIS	International Reactor Innovative and Secure
LRF	Large Release Frequency
LWR	Light Water Reactor
MM	Multi-modules
MZ	Monitoring Zone
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
PSA	Probabilistic Safety Assessment
PAZ	Precautionary Action Zone
PIE	Postulated Initiating Events
SMR	Small Modular Reactor
SPZ	Sanitary Protective Zone
SSTC NRS	Stave Scientific and Technical Center for Nuclear and Radiation Safety
TEDE	Total Effective Dose Equivalent

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UPZ	Urgent Protective Action Zone
VTT	Technical Research Centre of Finland
WP	Work Package

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1. Introduction

As discussed in INSAG-28, 'Application of the Principle of Defence in Depth in Nuclear Safety to Small Modular Reactors' by IAEA in 2024, small modular reactors (SMR) might be more prone to common cause failures, especially with respect to external hazards at the same site. With SMRs, the consideration of the multi-unit/multi-module aspects is becoming more important, because in many concepts there are more interactions and dependencies between the units (modules) than typical for current multi-unit sites.

There is distinction between multi-unit and multi-module SMR configurations. Multi-unit refers to several independent reactor units co-located on the same site, each with its own systems and potentially operating autonomously. Multi-module refers to multiple reactor modules integrated within a shared infrastructure, often with common systems, designed for operational and economic optimization. Sharing of some safety related systems and components (such as the control room, reactor pool and spent fuel pool) is the factor to consider. Furthermore, in order to preserve their modular attribute (regarding their source term), more than one module failing because of an external hazard (or any other common cause) would lead to unacceptable off-site consequences if the regulatory limits were set for a single module.

The small power output of individual SMRs may result in a lower source term (compared to large reactors) to be considered when defining the emergency planning zone. However, for multiple-modules sites, the total source term must be carefully assessed, as it may offset the expected advantages for emergency planning. Apart from this, the general requirements for defining emergency planning zones remain largely consistent with those for other types of nuclear facilities.

The aim of the deliverable is to evaluate aspects related to definition of emergency planning zone (EPZ) for multi-modules sites, and to provide recommendations regarding the calculation of the EPZ size. The EPZ is associated with the fifth and final level of defence in depth. The practical application of the fifth level of defence in depth should be guided by the outcomes of a plant-specific hazard assessment. However, the absence of a globally harmonized approach to defining EPZ and related requirements poses challenges to this process. Establishing consistent practices would be beneficial, particularly in such areas, as accident selection and source term evaluation, dose criteria, and the delineation of emergency planning zones.

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2. Definition of EPZ

2.1. IAEA recommendations

EPZs are areas identified around an NPP wherein all sorts of technical, logistical, and infrastructural arrangements are made to ensure timely and effective response in the event of an off-site emergency,

Requirements in relation to emergency planning zones and distances are provided in the IAEA GSR Part 7 and addressed in the associated lower level EPR publications. The IAEA GSR Part 7 defines four different regions around a nuclear power plant for which emergency response planning procedures have to be prepared in advance. 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 deterministic effects can occur. In this area 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 are the Extended Planning Distance (EPD) 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 are made to protect, monitor and analyse food and water supply.

Description of these zones is presented below, see Figure 1:

- Precautionary action zone (PAZ). PAZ is 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). UPZ is 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.

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– 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.

The suggested sizes for emergency planning zones and distances are provided in Table 1.3 of IAEA-EPR-NPP Public Protective Actions (2013) and Figure 1.

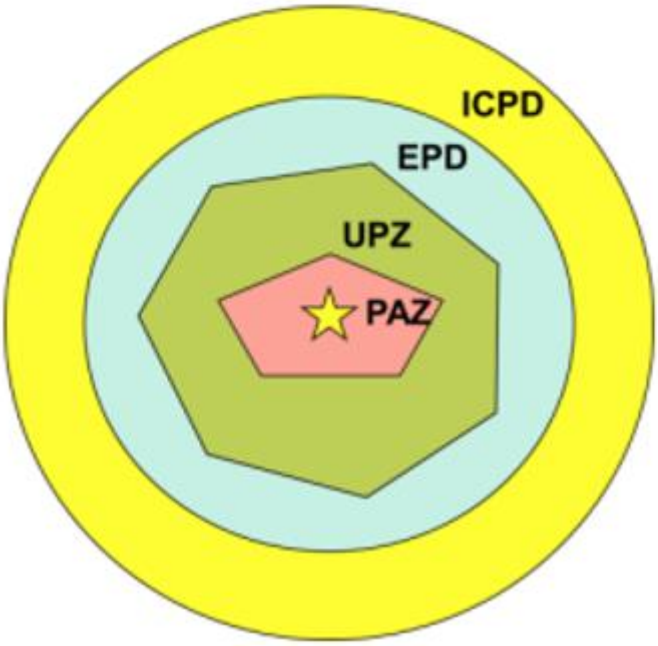

Figure 1. Emergency planning zones

The suggested sizes for emergency planning zones and distances applicable for nuclear power plants in general with thermal powers ranged between 100-1000 MWth) are

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provided in Table 1, extract from IAEA-EPR-NPP Public Protective Actions (2013). These distances are generic, namely they have not been computed by making plant-specific, design-specific calculations. The zones are not limited with national borders and may cross the territory of neighboring states (IAEA GSR Part 7).

Table 1. Suggested sizes for emergency planning zones

Emergency zones and distances	Suggested maximum radius
precautionary action zone	3÷5 km
urgent protective action planning zone	15÷30 km
extended planning distance	50 km
ingestion and commodities planning distance	100 km

The suggested size for the PAZ were proposed considering the following, : except for the most severe emergencies, 5 km is the limit to which early deaths are postulated; it provides about a factor of ten reduction in dose compared to the dose on the site; it is very unlikely that urgent protective actions will be warranted at a significant distance beyond this radial distance; it is considered the practical limit of the distance to which substantial sheltering or evacuation can be promptly implemented before or shortly after a release; and implementing precautionary urgent protective actions to a larger radius may reduce the effectiveness of the action for the people near the site who are at the greatest risk.
Illustration of relationship between magnitude of the releases, EPZ sizes and time, applied by IAEA approach, is shown on Figure 2.

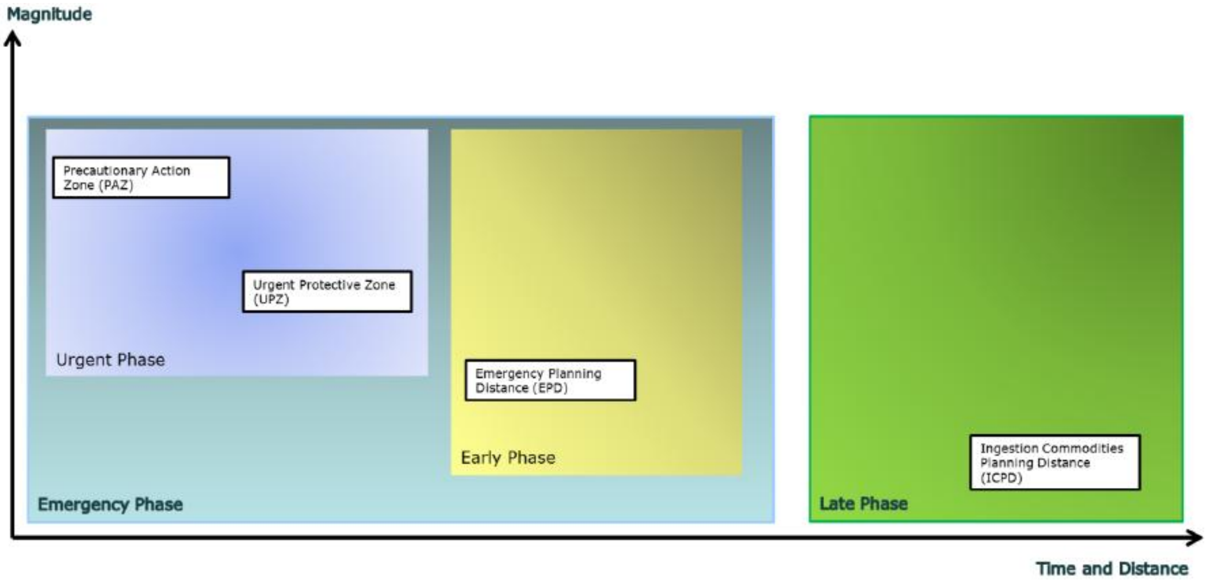

Figure 2. Conceptual drawing of Emergency Planning Zones, ELSMOR (2021).

Specific aspects on EPZ sizing and geometry are further provided in IAEA GS-G 2.1, which specifies the on-site areas, PAZ and UPZ (similar to IAEA GSR part 7). The PAZ and UPZ should be roughly circular areas around the facility and their boundaries should be defined by landmarks, where appropriate, such as rivers, roads etc. for easy identification during a response, cf. Figure 3. The sizing of the PAZ and UPZ is in accordance with the guidance for reactors with power from 100 to 1000 MWt_{th}:

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- PAZ radius 0,5 – 3 km
- UPZ radius 5 – 30 km.

Concerning the actions in such zones, the protective actions should include sheltering (max 2 days), iodine prophylaxis and evacuation. For long term protective measures, foodstuff restriction and relocation are considered.

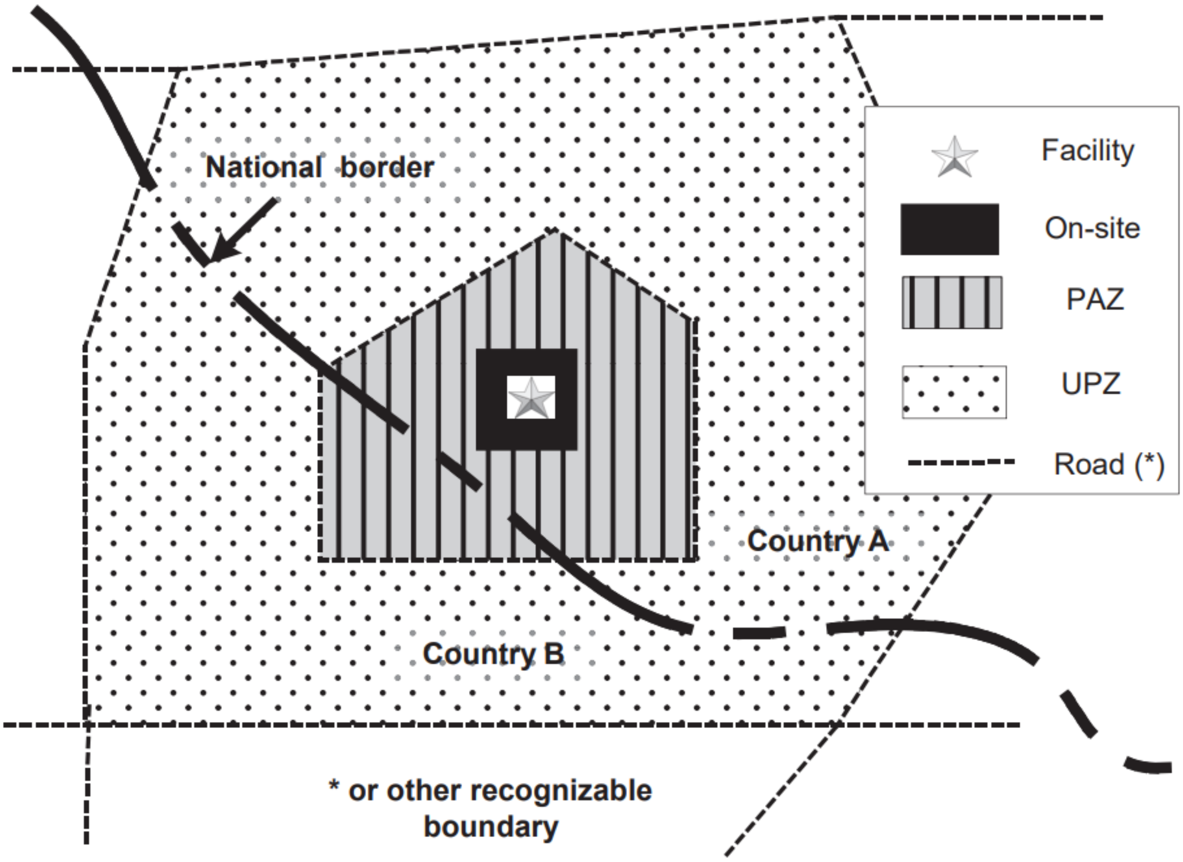

Figure 3. Emergency planning zones, IAEA GS-G-2.1.

More aspects regarding the SMR have been discussed at The SMR Regulators' Forum Emergency Planning Zone Working Group, established by IAEA in 2015 to identify, understand and address key regulatory challenges with respect to EPZ sizes that may emerge in future SMRs regulatory activities.
It was found in, 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 limited by the site boundary may be considered. Discussion of different design aspects that may influence on such statement is presented in Section 5 of SMR Regulators' Forum (2018). The list of aspects includes: small reactors and low rated thermal power levels; containment or containment function; subterranean location; novel features and technologies.
Regarding the EPZ determination methods, the following key considerations should be taken into account to determine the right-sized EPZ for SMR or any other NPP:

- Hazard assessment, also very low probability and beyond design basis accidents;
- Atmospheric source term, including timing;
- Public offsite doses;

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- Generic criteria (doses) for response actions;
- Effectiveness of response actions;
- Available resources;
- Integration into overall protection strategy;
- Adaptation to national & local circumstances;
- Optimization.

Overview of current methods for EPZ definition, with emphasis on initiating events, source term and offsite doses is shown in Ilvonen (2022): scaling by thermal power, provable exclusion of large releases/ practically eliminated sequences (theoretically allow to rid of the EPZ and rely on the site boundary only); full-scope PSA Level 3.

2.2.WENRA recommendations

The EPZ is tightly related to severe accident scenarios. The WENRA (2010) defines requirements on Accidents with core melt (O3), which requires reduction of potential radioactive releases to the environment from accidents with core melt, also in long-term, with following criteria:

- Accidents with core melt which would lead to early or large releases must be practically eliminated.

The latter criterium is even more stringent:

- for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures.

Safety of the new designs is described in the WENRA (2013), where to achieve the Objective O3 (in previous paragraph) on the 4th level of the defense in depth, following interpretations of limited protective measures are provided:

1. Immediate vicinity of the plant: For new reactors, the design should be such that the possible release of radioactive substances in a postulated core melt accident, based on the analysed consequences of the accident, will not initiate a need for emergency evacuation beyond the immediate vicinity of the plant. The design goal should aim at having a radius of this immediate vicinity zone towards the lower end of the suggested PAZ range i.e. **3 km (evacuation zone)**.
2. Limited sheltering and iodine prophylaxis: For new reactors, the design goal should be such that the possible release of radioactive substances in a postulated core melt accident, based on the analysed consequences of the accident, will not initiate a need for sheltering and iodine prophylaxis beyond the zone towards the lower end of the suggested UPZ range i.e. **5 km (sheltering zone)**.
3. No long-term restrictions in food consumption: This is interpreted so that after a postulated core melt accident, based on the analysed consequences of the accident, agricultural products beyond the sheltering zone should generally be consumable after the first year following the accident.

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4. Sufficient time: Sufficient time is interpreted so that protective measures should be initiated early enough. Especially the evacuation shall be carried out already when there is a threat of a significant radioactive release into the environment. Sufficient time to implement these protective measures is different for each measure and for each accident scenario and depends on the location of the reactor. Sufficient time for each measure shall be estimated and considered in the design of a reactor and during the site licensing.

The interpretation of limited protective measures of evacuation sheltering and iodine prophylaxis to be applied as goals in the design phase of new reactors is summarized in Figure 4. The EPZ zones are generally larger, because they are based on a conservative approach to protect people and considering the plant location and population living nearby.

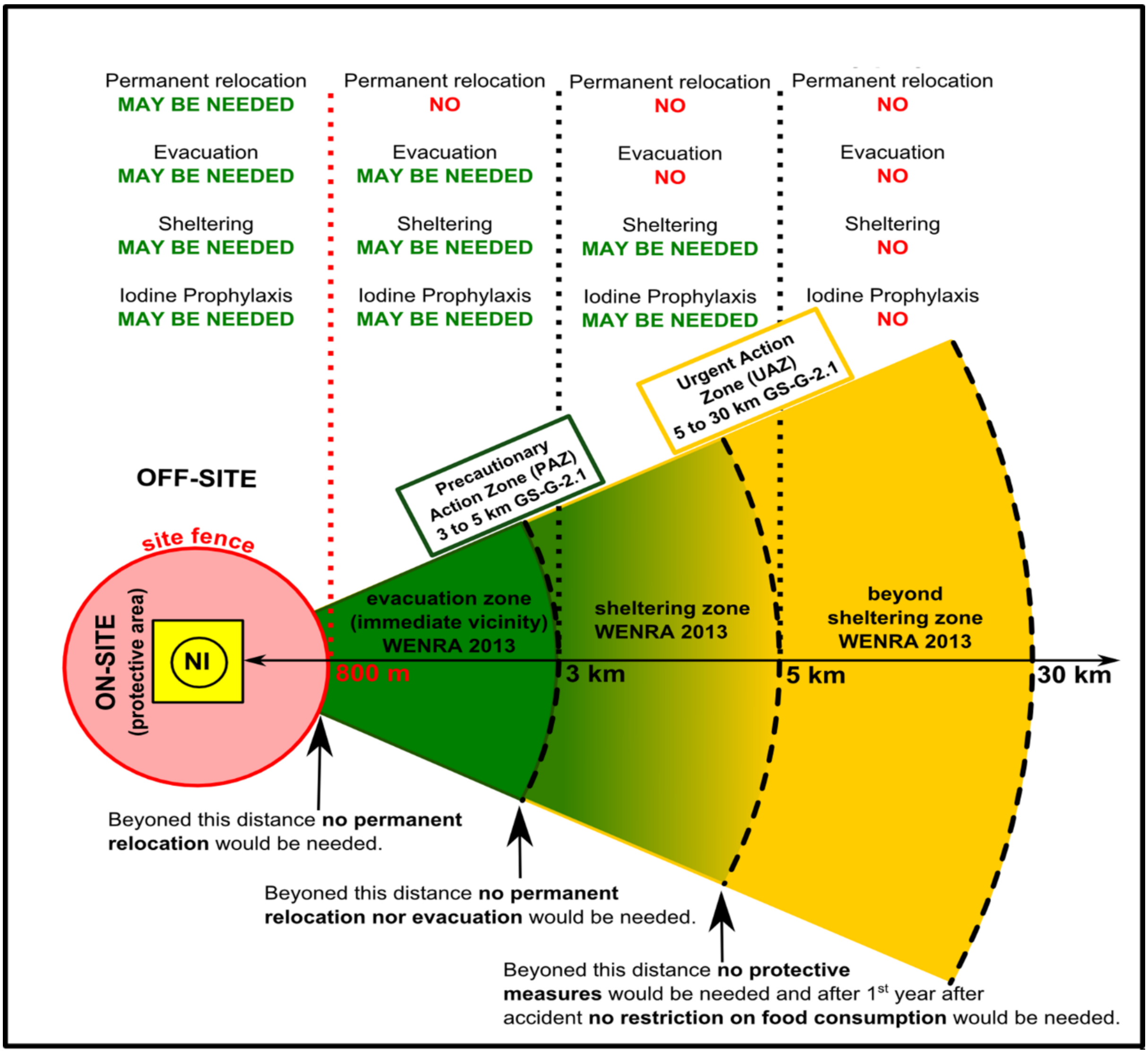

Figure 4. Design goals for areas where limited protective measures may be needed, WENRA (2013).

Specific requirements on SMR EPZ and O3 objective can be found in WENRA RWHG Report Applicability of the Safety Objectives to SMRs. According to the report, if SMRs are to be deployed in areas with relatively high density of population, since several designers claim that no EPZ is needed for their SMR concept (often called “EPZ on the

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fence"), more stringent acceptance criteria than those for O3 could be required by national Safety Authorities.

2.3. National regulations

2.3.1. Emergency planning zone practices

Current emergency planning zone practices varies from country to country. Table 2 represents an overview of EPZ sizes for different countries, compiled from Ilvonen (2022), Soni et al (2024), Locatelly and Mancini (2011), SMR regulator's forum (2018).

Table 2 Selected EPZ sizes for different countries

Country	**EPZ radius and details of urgent protective action**
Australia	Zone 1: 500 m (evacuation zone), Zone 2: 2.2 km (based on existing conditions) and exclusion zone: 1.6 km
Belgium	Evacuation zone: 10 km, sheltering zone: 10 km, Iodine thyroid blocking: 20 km
Canada	Evacuation zone: 7 km, sheltering zone: 10 km, Iodine zone: 10 km
Czech Republic	NPP Dukovany: 10 km evacuation zone, 20 km sheltering and stable iodine zone NPP Temelin: 5 km evacuation zone, 13 km sheltering and stable iodine zone
Finland	Emergency preparedness zone up to 20 km and iodine thyroid blocking, sheltering, evacuation up to for 5 km zone
France	Evacuation: 5 km, EPZ (sheltering and Iodine thyroid blocking): 20 km
Germany	Central zone: Surrounds the nuclear facility in a 2-km radius Intermediate zone: up to 10 km; outer zone: About 25 km
Hungary	PAZ up to 3 km, UPZ up to 30 km and long-term protective action planning zone up -71 km
Japan	Sheltering zone and evacuation zone up to 10 km (note: In Fukushima accident, emergency planning zone around the plant was within 10-km radius. Evacuation was planned within a radius of 2–3 km which was expanded up to 10 km and further extended up to 20 km from the plant in 2 days)
Luxembourg	Iodine thyroid blocking: Up to 25 km, distance for evacuation and sheltering are changed case by case
Netherlands	Evacuation zone: 5 km, Iodine thyroid blocking: 10 km, sheltering zone: Up to 20 km
Slovakia	Internal zone: 3 km for Bohunice Inner emergency zone: up to 12-15 km in radius around the NPP Slovakia Indication zone: up to approximately 50 km in radius around the NPP EPZ: 30 km Bohunice, 20 km Mochovche (divided into zones of 5 and 10 km)
South Africa	Internal zone: 5 km, UPZ: 5–16 km and long-term protective action planning zone: 80 km
Sweden	Inner emergency zone: Up to 12–15 km and indication zone: Up to 50 km approximately (note: during Chernobyl accident control on food stuff were required for up to 300 km to avoid detectable excess child thyroid cancers)
Switzerland	Internal zone: 3–5 km and sheltering zone up to 20 km
United Kingdom	Detailed emergency planning zone ranges from 1 to 5 km
USA	Two EPZs around each NPP. First, the plume exposure pathway EPZ whose radius is about 16 km and second, the ingestion exposure pathway EPZ whose radius is about 80 km from the reactor (note: in Three Mile Island nuclear

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Country	EPZ radius and details of urgent protective action
	accident evacuation was planned up to 5 miles and indoor sheltering was recommended within 10 miles (16 km). The evacuation zone was planned to further expand up to 10-mile (16-km) and then a 20-mile (32-km))
India	PAZ can extend up to 5 km, UPZ is 16 km and can extend up to 22 km and LPZ extend up to 30 km
China	Inner EPZ: up to 5 km, Outer EPZ: up to 10 km, Ingestion emergency planning zone: up to 50 km

According to Table 2, current EPZs are extremely different from country to country. The tendency is to establish large EPZs, if large reactors are employed and small EPZs for small reactors (see also, Table 3). There are no countries using small reactors that have large EPZ, but there are some (France and Germany) that have small EPZ even though all reactors are large.

Table 3. Proposed emergency planning zone dimensions of downsized reactor/small modular reactors designs, based on Kelk, et al (2020)

SMR design	Thermal power, MWth	EPZ size, km
BWRX-300	870	1
NUSCALE	200	0,5
Integral molten salt reactor	400	<0,5
Micro modular reactor	10	0,03-0,05
High Temperature Reactor-Pebble Module	250	0,5

2.3.2.European practice on EPZ sizing

Relevant information on European practice can be obtained from Jozef Kubanyi et al. (2008). This document includes information on existing EPZs of Belgium, Czech Republic, Finland, France, Hungary, The Netherlands, Slovakia, Switzerland and UK. Some of the countries EPZs are mentioned in the table above. The document mentions the so-called Environmental Impact Assessment (EIA), that is an assessment of the likely influence a project may have on the environment. The scope of the EIA procedure is to ensure that the decision makers consider environmental impacts before deciding whether to proceed with new projects. The EIA has little practical relevance to the issue of evaluating EPZ as there is no background technical guideline. Nevertheless, the issue of zoning is more and more mentioned in some current EIA studies for NPPs under operation.

A review of EPZ sizing for sheltering and evacuation in European scope is provided in document European Atomic Energy Community (2013). In general, the EPZ for sheltering vary from few kilometres to 30 kilometres, where majority falls within range of 10 to 30 kilometres. Only United Kingdom and Armenia have smaller zones, see

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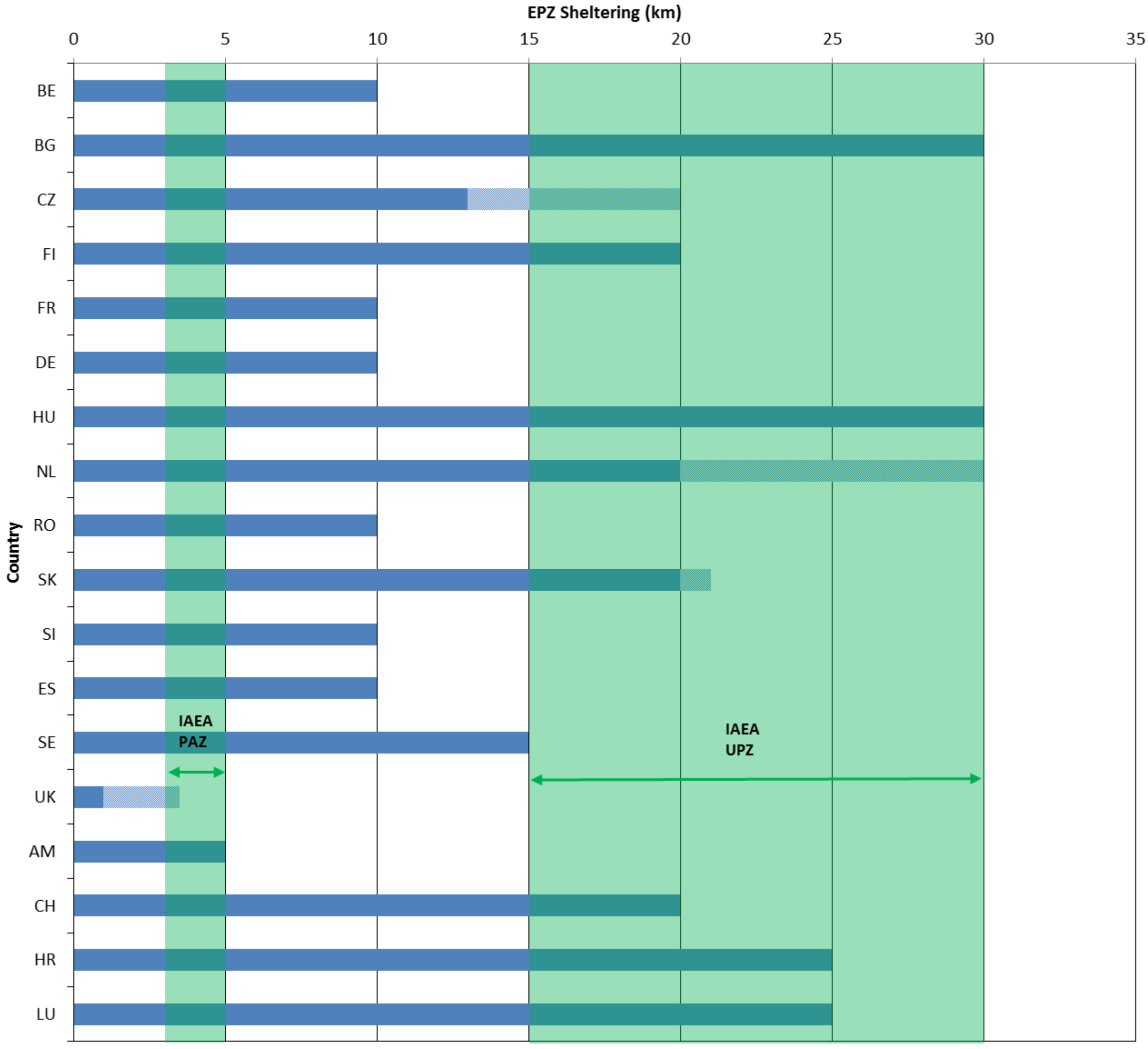
Figure 5. EPZs for evacuation vary from few kilometres up to 30 kilometres. Two thirds of the countries have EPZs for evacuation 10 kilometres or less, see Figure 6.



Notes:
BE - for rapid kinetic accidents, an intervention zone of 3.5 km has been set for NPP for immediate implementation of pre-determined actions
CZ - EPZ of 13 and 20 km for different nuclear sites
NL - EPZ of 20, 40 and 50 km for different nuclear sites (in NL, BE and DE, although in each case EPZ in NL is no more than 30 km)
SK - EPZ of 20 and 21 km for different nuclear sites
UK - detailed EPZ different for each nuclear site in range from 1 to 3.5 km; outline EPZ about 15 km
CH - 'zone 1' extends to between 3 and 5 km; 'zone 2' out to 20 km
SE - primary EPZ shown; also have secondary EPZ from 15-50 km, but this does not meet IAEA requirements for UPZ
IAEA - suggested maximum radii for zones for taking urgent protective action - PAZ from 3 to 5 km and UPZ from 15 to 30 km (NPP > 1 GW(th))

Figure 5. EPZ sizing for sheltering, European Atomic Energy Community (2013).

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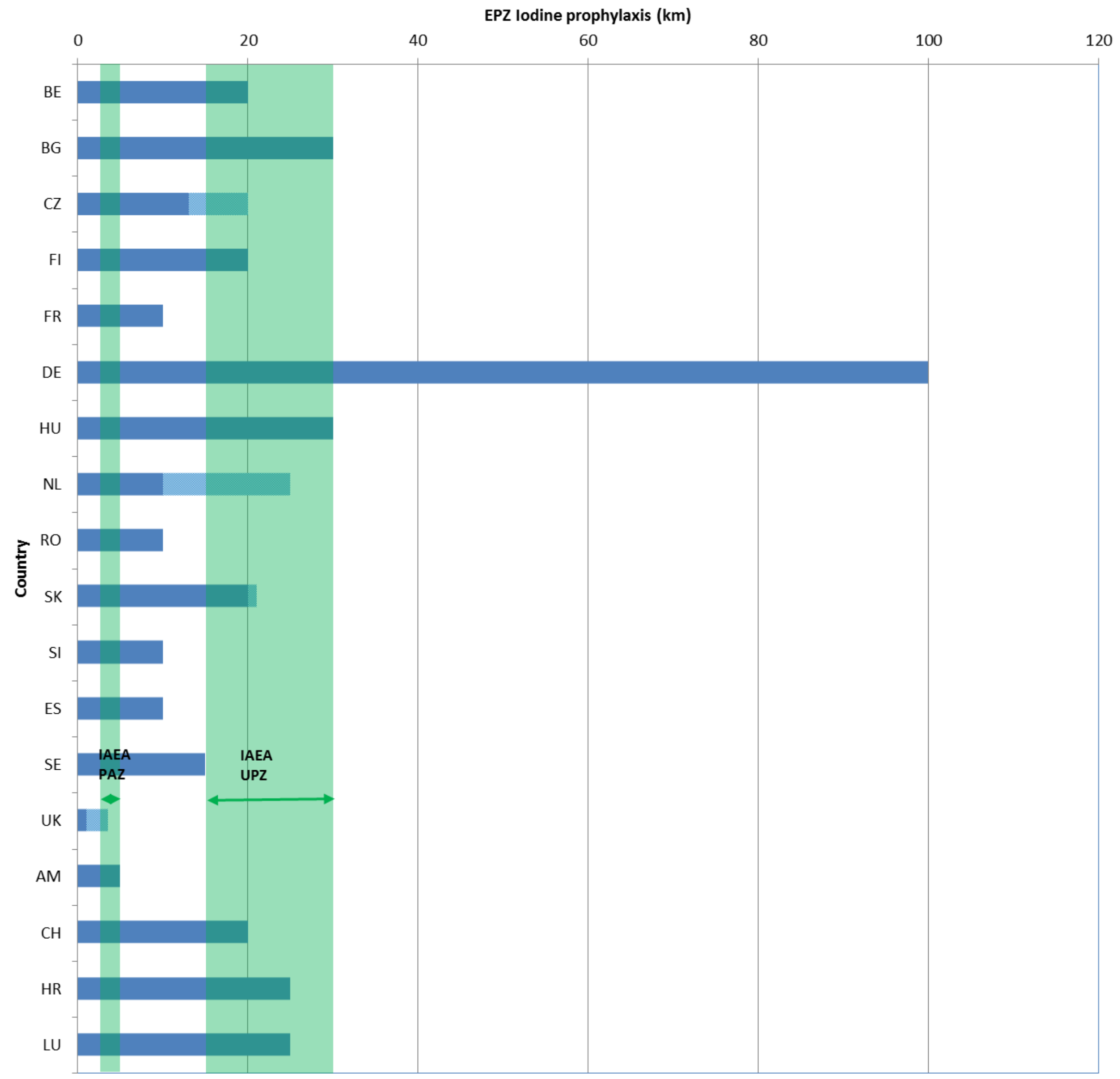

Notes:
CZ - EPZ of 13 and 20 km for different nuclear sites
NL - EPZ of 10, 20 and 25 km for different nuclear sites (in NL, DE and BE)
SK - EPZ of 20 and 21 km for different nuclear sites
UK - Detailed EPZ (DEPZ) different for each nuclear site in range from 1 to 3.5 km; outline EPZ about 15 km
CH - 'zone 1' extends to between 3 and 5 km; 'zone 2' out to 20 km
SE - primary EPZ shown; also have secondary EPZ from 15-50 km, but this does not meet IAEA requirements for UPZ
IAEA - suggested maximum radii for zones for taking urgent protective action- PAZ from 3 to 5 km and UPZ from 15 to 30 km (NPP > 1 GW(th))

Figure 6. EPZ sizing for evacuation, European Atomic Energy Community (2013).

2.3.3. Ukrainian regulations on EPZ-sizing

One of the basic principles of radiation protection, defined in Article 4 of the Law of Ukraine "On Nuclear Energy Use and Radiation Safety" requires that the individual doses, the number of exposed persons, and the probability of exposure from any of the types of ionizing radiation should be the lowest of those that can be practically achieved, taking into account economic and social factors.

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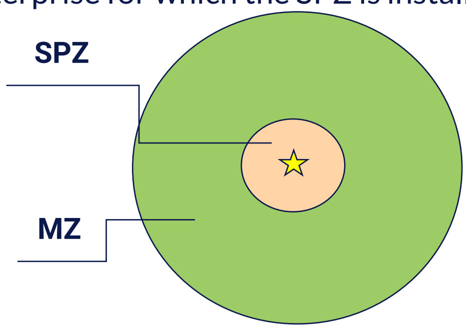
Based on this principle, a concept of EPZs is introduced in Ukraine: according to Article 45 of Law of Ukraine "On Nuclear Energy Use and Radiation Safety", in the place of location of a nuclear facility or radioactive waste management facility there shall be installed a sanitary protective zone (SPZ) and a monitoring zone (MZ).
This concept is in line with the approaches used in an international practice. However, up to now in a national legislative base there are no requirements for planning distances. It should be mentioned, that at the moment the process of harmonization of national general criteria with the existing IAEA standards are in progress.
NRBU-97 "Radiation safety standards of Ukraine" gives following definitions of SPZ (as PAZ supplemented by sanitary protection functions) and MZ (as UPZ):

- SPZ is a territory around the radiation-nuclear facility, where the level of public exposure in conditions of normal operation may exceed the dose limit. In SPZ the residence of the public is prohibited, there introduced restrictions on industrial activity, which is not related to the radiation-nuclear object, and radiation control is carried out;
- MZ is a territory on which impact of radioactive discharges and releases from a radioactive-nuclear facility is possible and where technological processes are monitored to ensure radiation safety of a radioactive-nuclear facility.

Traditional EPZs (SPZ and MZ) according to Ukrainian regulations with suggested sizes are shown in Figure 7. According to Article 45 of Law of Ukraine "On Nuclear Energy Use and Radiation Safety" and to OSPU-2005 "Basic sanitary rules for ensuring radiation safety in Ukraine", it is prohibited to place in the SPZ residential and public buildings, child care facilities, medical and health-improving institutions, facilities of economic and drinking water supply, industrial main and auxiliary buildings, which do not belong to the enterprise for which the SPZ is installed.



	Power 1375 and 3000 MW(th.)	
sanitary protective zone	SPZ	2.5 km
monitoring zone	MZ	30 km

Figure 7. Sanitary protective zone and monitoring zone according to Ukrainian regulation

According to the current legislative and regulatory framework, in particular, Law of Ukraine "On Nuclear Energy Use and Radiation Safety", OSPU-2005, NP 306.2.245-2024, the size of SPZ must be justified in the design documentation of the enterprises with radiation-nuclear technologies.
Article 33 of the Law of Ukraine "On Nuclear Energy Use and Radiation Safety" and Article 1 of the Law of Ukraine "On Authorizing Activities in the Sphere of Nuclear Energy Use", state that the legal entity conducting the activity related to the design of the nuclear facility is the operating organization (the Operator).

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This means, that if it is necessary to define or review the size of EPZs, it is the Operator, who is responsible for making calculations, substantiating their results and further agreeing with the state sanitary and epidemiological service institutions.
With regard to definitions of EPZs, Ukrainian regulatory framework does not differentiate the type of radiation-nuclear facilities, so, it is expected that current provisions in force will also apply to SMRs.
According to Minutes of the meeting (2024) of joint working group on the development of approaches and algorithms for determining the size of EPZs, emergency planning zones and distances for operating Ukrainian nuclear power plants (NPPs) can be appropriated in term of IAEA GSR Part 7 and IAEA-EPR-NPP Public Protective Actions (2013). The procedure for establishing emergency planning zones for new designs is not implemented.

2.4. Compliance criteria

2.4.1. Ukrainian intervention levels and other dose assessment criteria

Ukrainian national radiation safety standards containing the key dose limits and intervention levels do not specify any additional requirements to multi-unit or multi-module nuclear facilities. Its use the term "facility" as a single object that should be regulated by particular dose limits. However, according to Ukrainian framework, impact of multi-modularity potentially touches to frequency of severe accident sequences and the magnitude of the merged source term, and does not provide any specific requirements on multi-module cases in light of EPZ-sizing.
In order to ensure radiation protection of the public, dose limit quotas are established for exposure from all airborne and liquid discharges during normal operation and AOOs: there are 80 μSv/year for nuclear facilities and 40 μSv/year for radwaste management facilities in operation.
Separately, for airborne discharges for NPPs dose limit quota 40 μSv/year is established for the events with frequency higher than 1E-2 1/year (normal operation and AOOs). Based on the dose limit quota for each individual facility, permissible releases and discharges are determined, which are not allowed to be exceeded.
Radiation exposure for the public in case of DBAs and Design Extension Conditions (DECs) are limited through protective action taken in compliance with intervention levels and operational intervention levels. Intervention levels are determined in terms of averted dose due to protective actions. NRBU-97 established criteria (intervention levels and action levels) to make decisions on justification or unconditional justification of protective actions.
The protective actions and criteria for their implementations are laid down by the national radiation safety standard NRBU-97, according to which countermeasures are divided into immediate, urgent and long-term as specified below:

- immediate protective actions include protective actions aimed at preventing such levels of acute and / or chronic exposure of the general public as to the risk of clinically apparent radiation effects;

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- protective actions are classified as urgent if their implementation is aimed at preventing deterministic effects;
- long-term measures include protective actions aimed at preventing doses of short-term or chronic exposure, the values of which are usually below the thresholds for inducing deterministic effects.
- Immediate and urgent protective actions at the early phase of the accident include:
- sheltering;
- evacuation;
- restrictions in the mode of behaviour (limitation of time spent out-of-doors);
- iodine thyroid blocking (iodine prophylaxis);
- temporary ban on the consumption of certain food products of local production and the use of water from local sources.

Long-term protective actions can be applied at both the early and the late phases of the accident and include:

- relocation (resettlement to a permanent place of residence);
- temporary resettlement;
- restrictions on the consumption of radioactively contaminated water and food;
- decontamination of territories;
- agricultural protective actions;
- other protective actions (hydrological, including flood control measures, restrictions on forest use, hunting, fishing, etc.).

Criteria for immediate protective actions are given in Tables D.6.1, D.6.2, and D.7.1 of NRBU-97. Criteria for long-term protective actions are given below in Table D.8.1-8.3 of NRBU-97. The intervention levels and other dose limits are used as an offsite dose assessment criterion for releases and discharges in cases of normal operation of the nuclear facility, AOOs, DBAs and DECs with no reference to multi-unit/multi-module nature of the releases or discharges. Therefore, the specific features of multi-unit/multi-module is expected to be reflected in the probabilistic characteristic of the event only.

2.4.2. Requirements to frequency of the event considered in EPZ-sizing

Deterministic methods are used to define safety margins relative to all natural hazards including severe weather conditions. Probabilistic safety analyses are developed to determine risk contributors resulting from external impacts and other initiating events, identify safety deficiencies and develop additional compensatory measures. At the same time the probabilistic metrics such as large release frequency can be used to select the representative events to be considered under EPZ-sizing.
According to new document “General Safety Provisions for Nuclear Power Plants” (NP 306.2.245-2024), the safety criteria for NPPs for which the license for the right to carry

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out activities at the life cycle stage "operation of a nuclear facility" was issued before the entry into force of the Order of the State Nuclear Regulatory Inspectorate of Ukraine No. 195 dated March 4, 2024 are:

– not exceeding the value of the frequency of severe core damage calculated for the full range of initial events in all operational states of the power unit by 1E-4 per reactor per year. It is necessary to strive to ensure that the value of the frequency of such core damage does not exceed 1E-5 per reactor per year;

– not exceeding the value of the integral frequency of the maximum accidental release of radioactive substances into the environment of 1E-5 per power unit per year. It is necessary to strive to ensure that the frequency of such an accidental release does not exceed 1E-6 per reactor per year.

The safety criteria for NPPs for which the license for the right to carry out activities at the life cycle stage "construction and commissioning of a nuclear facility" was not issued before the entry into force of the Order of the State Nuclear Regulatory Inspectorate of Ukraine No. 195 dated March 4, 2024 are

not exceeding the value of the frequency of severe core damage calculated for the full range of initial events in all operational states of the power unit by 1E-6 per reactor per year;

not exceeding the value of the integral frequency of the large release into the environment of 1E-7 per reactor per year.

The size of the MZ is determined so that in case of DECs, the frequency of which is equal to or exceeds the above mentioned Large Release Frequency (LRF) values, the doses to the public on the internal boundary of the MZ and beyond it do not exceed the criteria for urgent protective actions (lower levels of justification) – evacuation and iodine prophylaxis according to NRBU-97.

Ukrainian regulatory framework has no specific requirements to safety metrics of multi-unite/multi-module events.

The lowest value of LRF 1E-7 per reactor per year is also considered as a boundary of frequency spectrum to take into account some accident sequences in development of EPZs for SMRs (SMR Regulators' Forum, 2018). It is useful limitation in selection of accident scenarios in risk-informed approach. However, the value is not a rule for emergency preparedness goals. E.g., some severe accident scenarios can be assumed as a representative in EPZ-sizing independently from PSA level 2 results.

2.5. Multi-module SMR-specific considerations in EPZ-sizing

One of the key international initiatives for EPZ-sizing for SMRs is the EU SASPAM-SA project (2024), launched in 2022. Its main objective is to assess the applicability of existing knowledge from operating large Light Water Reactors (LWRs) to the near-term deployment of integral Pressurized Water Reactors (iPWRs), with a focus on severe accident analysis and EPZ requirements for European licensing.

Work Package 6 (WP6) of the project SASPAM-SA addresses the evaluation of EPZ size for selected severe accident scenarios, by coupling best-estimate severe accident codes with radiological consequence analysis tools. A notable feature of iPWRs is their

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potential siting in or near densely populated areas, attributed to reduced offsite consequences and enhanced inherent safety. WP6 aims to provide a scientifically sound basis for EPZ assessment. However, the project only addresses single-module events and does not consider multi-module configurations in its EPZ analysis.

According to the IAEA, SMRs are defined as advanced nuclear reactors:

- power capacity of up to 300 MW(e) per unit;
- intended for commercial applications (electricity production, desalination, process heat);
- designed for modular deployment at a single site;
- possibly using light or non-light water cooling;
- typically implementing novel designs that have not been widely licensed or operationally validated.

The IAEA SMR Regulators' Forum and Coordinated Research Projects (e.g., Magruder, 2017; Vilar-Welter, 2018) have identified several key safety and regulatory considerations related to multi-module SMR plants, including:

- use of passive safety systems and slower accident progression;
- increased complexity due to multi-module human-machine interactions;
- shared control rooms and physical security at new sites;
- potential for common-cause failures;
- nonlinear relationship between source term magnitude and EPZ size;
- challenges in demonstrating "provable safety" due to innovative and untested design features.

The U.S. Nuclear Regulatory Commission (NRC) discussed regulatory aspects for multi-module SMRs as early as 2011. However, given the diversity of SMR technologies, no specific EPZ requirements exist. The only guidance appears in Standard Review Plan Section 19.0, which requires applicants to:

- systematically identify accident sequences, including those involving multiple modules and significant human errors;
- propose design features and operational strategies to prevent or mitigate those sequences;
- demonstrate that such sequences are not significant contributors to overall plant risk.

Although these requirements address multi-module risk, they do not establish quantitative criteria for EPZ sizing in such configurations.

Several studies, including VTT's Ilvonen (2022) EcoSMR report and the Kelk, R., Murad, A., de Oliveira, R., & Jeltsov, M. (2020), argue that SMRs may allow for significantly reduced EPZs, even down to the site boundary, due to:

- lower reactor power and fuel inventory per module;
- reduced source term and offsite release potential;

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- slower accident kinetics, giving operators more time to respond;
- each module having independent, full-scope safety systems;
- robust containment structures (e.g., compact, high-pressure-resistant, below-grade, water-immersed designs);
- exclusion of on-site refueling, reducing human intervention and on-site risk.

These technical characteristics make multi-module SMRs fundamentally different from traditional large LWRs in EPZ planning, and they support the development of right-sized, risk-informed EPZ frameworks. While existing NPPs may operate as multi-unit sites, they do not feature the integrated, co-located multi-module configurations envisioned for SMRs. The latter introduce specific challenges, such as the potential for common-cause impacts across modules within a shared reactor building, interdependencies of safety systems, and cumulative source terms in accident scenarios. These aspects distinguish multi-module SMRs from both single-unit LWRs and conventional multi-unit sites, and they highlight the need for harmonized regulatory approaches and quantitative methodologies tailored to multi-module configurations - a gap that remains in current international practice.

2.5.1. Dose-relevant radionuclides and representative source term

The main contributors to the dose can be predicted based on the maximum number of isotopes available in the neutron-physical code library. Moreover, for the representative of the group such as normal operation, AOOs, DBAs and severe accidents (DEC-B), the nuclide packages will be very different. In the case of severe accidents, the isotopic composition of the release source will strongly depend on the state of the facility under consideration and delay before the release.

In case of multi-unit releases, merged source term should cover all dose-relevant radionuclides including those that still relevant for all exposure pathways.

Proposed preliminary approach adopted to select the representative source terms involves the following main steps:

- review of the results of related tasks on inventories and release fractions;
- selection of a representative unit/module and spent fuel pool;
- development of an event tree to address the key factors affecting the release magnitude using a simplified approach and verification for certain branch of the event tree;
- estimating the source terms for dose-relevant nuclides based on the developed end-state tree;
- comparing the radiological equivalents of the obtained source terms;
- selection of representative release sources depending on the expected timing of events and the magnitude of source term.

These steps are especially important when selecting a dose-relevant radionuclides and source terms based on the results of severe accident analysis/PSA level 2. This approach

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allows an expert to define the concept of a representative power unit/module, which, *ceteris paribus*, allows an expert to extrapolate this approach to multi-unit/multi-module cases. An example of the extrapolation can be calculation of source term as n times the single unit source term for EPZ determination.

In practice, for new reactor design applications, the industry is using mechanistic models of fission product release and transport. In SECY-1-0012 (2016), the NRC staff stated that non-LWR applicants can use modern analysis tools to demonstrate quantitatively the safety features of new reactor designs. For example, one SMR vendor used the MELCOR code to estimate source terms for its safety analysis report. More recently, non-LWR reactor vendors have submitted topical reports describing their planned use of mechanistic models to estimate source terms.

In the NRC staff requirements memorandum to SECY-93-092, the Commission approved the NRC staff's recommendation that source terms for non-LWRs be based upon a mechanistic analysis and that the acceptability of an applicant's analysis will rely on the NRC staff's assurance that the following conditions are met:

- The performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on the reactor and fuel performance through the research, development, and testing programs to provide adequate confidence in the mechanistic approach.
- The transport of fission products can be adequately modelled for all barriers and pathways to the environment, including the specific consideration of containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured.
- The events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties.
- The design-specific source terms for each accident category would constitute one component for evaluating the acceptability of the design.

2.5.2.Near-range effects

The publication on dispersion modeling for SMR releases presented in frame of IAEA Conference book of abstracts (2023) demonstrates increasing interest in locating SMRs closer to potential end users for industrial or district heating applications, which is making it more important to understand the near-field atmospheric dispersion behaviour of routine or accidental radionuclide emissions. This work addresses these limitations and complements existing practices by using high-fidelity computational fluid dynamics (CFD) modeling for a realistic assessment of near-field radionuclide dispersion on a complex site.

The near-range effects of radionuclide dispersion from SMRs play a critical role in evaluating EPZs, especially due to their potential siting near population centers and unique design features. Studies such as the Ilvonen EcoSMR report (VTT) emphasize that SMR offsite doses could be significantly lower than those from large NPPs, often remaining below thresholds for acute health effects. This dose threshold characteristic

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— where no health effects occur below a certain dose level — is a key argument for reducing EPZ size around SMRs. These characteristics lead to stronger radionuclide concentration in near-field areas, particularly within the first kilometres from the release source. As a result, local atmospheric dispersion and dose behavior become highly variable and site-dependent, especially under accident scenarios. At the same time, claims of significantly reduced EPZ distances, or even EPZs coinciding with the site boundary, demand a higher level of demonstration. Such reductions imply that no protective actions would be required outside the site perimeter, which necessitates highly reliable, site-specific assessment methods rather than relying solely on conventional long-distance dispersion models. Moreover, given that SMRs may be located in more densely built environments, nearby structures and urban morphology can further modify atmospheric dispersion, adding complexity to dose evaluations and reinforcing the need for advanced, high-resolution modeling approaches.

The IAEA Conference (2023) highlighted growing interest in deploying SMRs closer to end users for industrial or district heating. This trend further underscores the importance of understanding near-field dispersion dynamics under both routine and accidental release conditions.

Recent near-range modeling, performed under sensitivity analysis of the offsite results of a single code to different effective heights and building wake effect, which were studied in detail using HotSpot code (under the EU SASPAM-SA project, 2024), demonstrate that in case of severe accident at single-module SMR (180MWth):

- most deviations in predicted dispersion related to an effective release height and building downwash effect occur within 1 km of the release point;
- standard Gaussian plume models often fail to capture detailed local effects, especially due to building downwash and terrain-induced airflow patterns;
- CFD modeling provides a higher-resolution alternative for site-specific dispersion analysis, but requires more computational resources;
- at distances beyond 1,000 meters, modeling predictions become more stable and less sensitive to initial conditions, with convergence observed across Gaussian plume, Gaussian puff, and Lagrangian particle models.

In continuous release or statistical assessment scenarios, building geometry (particularly asymmetry in width vs. length) and terrain configuration significantly affect radionuclide concentration distributions due to induced airflow asymmetries. These effects should be carefully studied during site-specific EPZ determination, especially for SMRs situated near urban or industrial areas.

Configuration of separated multi-unit structures onsite will be accompanied with effects described in Section 3.2.2.

2.5.3. Release pathways and configuration of the release points onsite

Understanding the release pathways and the configuration of discharge points on-site is essential for accurate evaluation of source terms, atmospheric dispersion modeling, and ultimately, EPZ determination for SMRs.

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In most SMR designs, radioactive effluents are managed using tightly controlled systems that minimize routine and accidental releases. For example, treated liquid effluents are normally recycled as demineralized water within the reactor island. If reuse is not possible—due to operational constraints (e.g., water balance control) or abnormal events (e.g., fuel failure)—these liquids are discharged through a single controlled outlet to the environment.

Similarly, gaseous effluents are managed by nuclear HVAC systems designed to:

- maintain negative pressure differentials to prevent uncontrolled leakage;
- control the airflow direction from low- to high-contamination zones;
- operate at nominal air exchange rates (typically 1 to 4 air changes per hour) in safety-classified buildings such as containment;
- direct air to a common emission stack fitted with High Efficiency Particulate Air filtration units, as per recognized good practices, to remove airborne particulates before atmospheric release.

These systems define the spatial and temporal characteristics of releases under both routine and accident conditions.

Computational tools like RADTRAD, used for source term assessment in licensing and EPZ studies (e.g., for the IRIS SMR model), require detailed modeling of release pathways between compartments within the Nuclear Steam Supply System. In the IRIS configuration, 13 release pathways were identified between core components and boundary systems. Core components are components within the reactor pressure boundary that directly involve nuclear fuel and the primary coolant loop, being the initial source of any potential radioactive release during an accident. There are reactor core (fuel assemblies), reactor pressure vessel, team generators (if integral to the reactor, as in IRIS), etc. Boundary systems define the outer limits of the containment. They are the last barriers before a release to the outside world occurs.

There are containment vessel or reactor containment building, secondary containment (if applicable), main steam isolation valves, containment isolation system, filtered containment venting system, etc. These systems are designed to delay, filter, or minimize the release of radioactive material during an accident.

To be used effectively in EPZ analysis, source term descriptions should include:

- nuclide-specific release rates (Bq/s);
- time-dependent profiles of release intensity;
- vertical distribution of releases (i.e., release height, plume rise) at each time step.

Precise definition of these parameters — especially initial conditions prior to atmospheric dispersion — is essential for modeling near-field dose gradients and validating radiological impact assessments for SMR installations, particularly when sited close to highly-populated areas.

It should be noted that many SMR designs are built underground. Because of that, it is more likely to have an accident whose source term is released at a very low elevation. Such low height of emission has an impact on the source term diffusion and advection. In

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practice, the effective release height is going to be assumed to be minimal (ground discharge) or greater.

2.5.4. Merged source term

Merging two or more releases from a different release pathways onsite can be a non-trivial task in terms of multi-modular events. In the case of CFD modeling, the use of atmospheric dispersion models may require additional information on the topography and spatial resolution of release points.

Special attention should be paid not only to the spatial resolution, but also to the time resolution of the release source. The source term kinetic is continuously linked to the meteorological resolution used to model atmospheric dispersion. These two related modeling options should be well harmonized, possibly requiring additional justification of their selection due to sensitivity analysis.

In the context of multi-module SMRs, defining a merged source term — i.e., the combined release from multiple modules or multiple release pathways — is a complex but essential step in assessing radiological consequences, modeling dispersion, and EPZ-sizing.

The merging of source terms across modules or pathways is not a straightforward summation. It requires consideration of:

– temporal synchronization of releases (timing, duration);

– spatial configuration of release points (relative locations, elevation).

Another aspect is modelling. In case of using CFD-modeling. CFD-based atmospheric dispersion modeling requires high spatial and temporal resolution of release characteristics. The source term kinetics (i.e., how activity changes over time) must align with the meteorological resolution used in dispersion models. Inconsistent resolutions may result in modeling errors or underestimated peak doses. Therefore, harmonizing these parameters is crucial, and often requires sensitivity analysis and justification of assumptions used in source term selection.

The temporal resolution of several source term should be sufficient to reflect important phenomena affecting release intensity. For multiple sources, it is recommended to select a single time reference system to enable the combination/aggregation of activity or any other characteristics from multiple modules/units into a single merged source term.

2.5.5. Temporal resolution of source term

Time-resolution of source term for single-unit as well as multi-unit/multi-module cases plays significant role in receiving accurate and reliable results of dispersion modeling.

According to study on atmospheric dispersion modeling uncertainties “Guidelines for ranking uncertainties in atmospheric dispersion” provided by European Joint Programme for the Integration of Radiation Protection Research (EJP-CONCERT, 2018) demonstrates that source term and meteorology are the main factor contributing to uncertainties in atmospheric dispersion modeling and dose assessment. Hence, temporal resolution of the source term — i.e., the time-dependent characterization of radionuclide release — is a critical input parameter in modeling atmospheric dispersion and assessing radiological impact in frame of EPZ-sizing.

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The actual simplified approaches to dispersion modelling using Gaussian plume models do not require a high level of precision on temporal resolution of source term. Sum of activity by all time-intervals can satisfy the condition of assessment. On other hand, modern dispersion modeling tools, such as VALMA, JRODOS, ARGOS rely on synchronized time-resolution between source term data and meteorological inputs. For instance, RIMPUFF atmospheric dispersion model, when integrated with the JRODOS meteorological pre-processor, requires a minimum time step of 10 minutes to maintain sufficient sensitivity and accuracy in the near- and mid-range (up to 21 hours after release onset).

An accurate source term definition should include:

- nuclide-specific release rates (Bq/s);
- time-resolved release profiles, describing how emission intensity changes over time. Release rates can take the diverse values during the accident progression. It is important, especially, while the assumed weather is unstable;
- initial plume rise parameters (thermal power, vertical flux and geometry of nozzle) of released mixture at each time step.

Overall, choosing a sufficiently fine temporal resolution—typically in the range of minutes—is essential to:

- capture short-term variations in source strength;
- properly represent the kinetics of early-phase releases;
- ensure compatibility with high-resolution meteorological datasets;
- reduce uncertainty in dose estimation, especially in near-field EPZ analyses.

In multi-unit/multi-nodule release modeling, one of the main task is to receive the full release time-line from all sources, considering their further merging in frame of dispersion modeling. Time step should be harmonized among each source term and with a time step of applied meteorological data.

2.5.6.Meteorology and auxiliary data

One of the key specific of meteorology used in modeling radiological consequences of multi-unit/multi-module events is spatial resolution of gridded numerical weather data that should cover all possible release sources and release pathways onsite as well as related near-range effects (see Sections 2.5.2 and Section 3.2.2). General requirements to meteorology still the same as for single-unit assessments and its are compiled in Section 3.3.5

When it comes to analyzing the expected scope of protective actions including long-term measures, some additional offsite data may be required to assess long-term doses and protective measures:

- population density and demographic distributions;
- topography (elevation, slopes, drainage patterns);
- land use/land cover maps, soil type and composition;

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- hydrology including surface water (rivers, lakes, wetlands), groundwater flow and aquifers;
- evacuation roads, types of buildings/shelters and other emergency response infrastructure;
- types of crops grown and seasonal cycles, livestock presence and type, agricultural land area, irrigation practices and water sources;
- soil-to-plant transfer factors for radionuclides;
- food distribution and supply chains, food consumption patterns of local population.

In practice, most of data should be customized (for particular region where the site is located) and integrated in actual calculation tools.

2.5.7.Computational limitations

The suitability of various codes for atmospheric dispersion modeling and dose projection in determining EPZ sizes for iPWRs may be constrained by factors such as modeling scale, data availability (e.g., land use or agricultural production), meteorological conditions, spatial resolution, and software capabilities. A comparative analysis can assist users in selecting the most appropriate preliminary tool for assessments aligned with iPWR-specific requirements.
SSTC NRS has conducted a brief qualitative evaluation of commonly used tools in frame of EU SASPAM-SA project (2024) for a single module. The results provide an overview of the general characteristics and features of the HotSpot code, the JRODOS decision support system models, and the computational algorithm described in NP 306.2.173-2011. This initial review is intended to identify codes suitable for subsequent conservative and best-estimate calculations. Final conclusions and recommendations will follow after a series of simulations aimed at determining the EPZ dimensions for iPWRs.
However, at this stage, it can already be stated that a limitation of some software for modeling multi-unit events may be the backwardness of inputting source information for multiple sources on site. However, with the development of new technologies and GIS systems, the results of almost any atmospheric dispersion model obtained for a single unit can be transferred and combined with results for other power units.

2.5.8.Risk considerations

Depending on selected approach for EPZ definition, considerations on risk of core damage, as well radioactive release risk should be accounted.
MM core damage frequency and large early release frequencies can be estimated as the results of design PSA Level 1 and Level 2, or as bounding estimate on the conditional probability that multiple modules would experience core damage (or large release) following core damage (or large release) in a single module.
For probabilistic approaches (see Section 3.4), if a multi-module accident sequence frequency is less than 10-7 1/year, then it can be neglected in the determination process of EPZ (NEI, 2013).

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For some SMR designs, since the multi-module SMRs are designed more carefully to escape from the multiple events, it is possible to use the EPZ derived based on a single module for the multi-module SMRs. Thus, for the same capacity reactor (e.g., 1000 MW), the more module SMR (e.g., 10 modules of 100 MW) has shorter EPZ distance than the less module SMR (e.g., 4 modules of 250 MW), Kim (2021).

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3. EPZ calculation approaches

3.1. General overview

General approach for determination of EPZ size is presented in EPR-NPP Public Protective Actions (2013). Application of the approach for SMRs was proposed in SMR Regulators Forum (2018), see illustration on Figure 8.

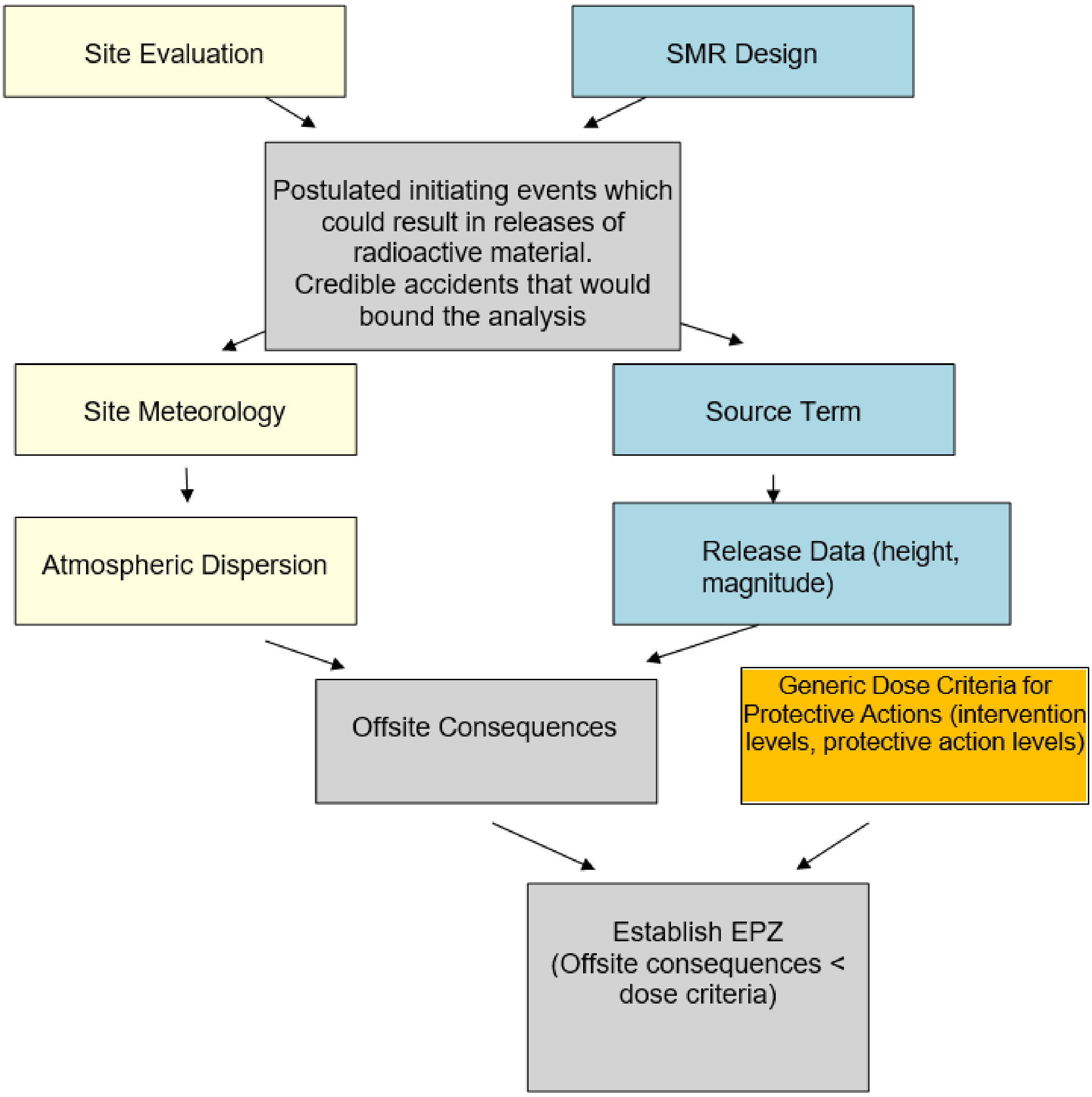

Figure 8. Flowchart to establish EPZ

Site evaluation step include consideration of several aspects important to emergency preparedness and emergency planning. It includes information on seismic, hydrological, geological, tidal, and other technically relevant subjects that support the site being suitable for the operation of a SMR; ability to decontaminate and to have long-term storage of spent fuel; physical protection of the site; population density; human

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assistance response means and times (fire, police, medical assistance); transportation routes (air, land, and waterways); sensitive environmental characteristics for cultural, biological, societal impacts.

The plant design should detail the planned number of operating reactors, power levels, electrical distribution, fuel characteristics, design means to prevent damage of fuel located in the reactor and/or in spent fuel storage, design means to prevent release of radioactive materials (source term) to environment, other design considerations.

Source term refers to the types and quantities of radioactive or hazardous materials released into the environment following an accident at NPP. It encompasses the magnitude, composition, physical and chemical form, and the mode of release (such as puff, intermittent, or continuous) of radioactive substances—primarily fission or activation products—emitted during a reactor incident. In addition, the mechanism, timing, and location of the release must be specified. The radiological consequences of such releases are generally categorized as follows:

- Within the reactor building, potentially exposing operational staff or personnel inside the structure.
- On-site but outside the reactor building, affecting other areas of the facility.
- Off-site impacts, potentially exposing the public through:
  - External exposure to released materials.
  - Internal exposure through inhalation or ingestion of radioactive substances.

Identification of postulated initiating events (PIEs) is required to define the scope of accident scenarios to be analyzed in the safety analysis. The range of PIEs must cover all credible accidents that could have an influence on the safety of the reactor. The scope of PIE should include internal events, internal and external hazards, as well combination of these events. 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.

The source term derived from the PIE should be combined with the site's meteorological data and appropriate dispersion modeling to evaluate the spread of radioactive isotopes. Once the atmospheric dispersion and ground deposition patterns are established, exposure pathways and radiation dose calculations can be conducted. These dose assessments are then compared to predefined dose criteria used for emergency response planning. Based on this comparison, the necessary distances for implementing protective actions—such as sheltering or evacuation—are determined.

Several methods can be applied to identify PIE, define the source term and to establish the size of EPZ. They can be classified as follows:

- Scaling approaches, see section 3.2;

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- Deterministic approaches, see section 3.3;
- Probabilistic /risk-informed approaches, see section 3.4.

Regarding the MM SMRs, the following point can be addressed for evaluation of source term for EPZ definition:

- Evaluation of source term from multi-module accident. It implies that probabilistic approach should be applied, and multi-module PSA Level 3 should be developed;
- Use sum of source terms from all single modules located at MM site. Scaling approach or deterministic approach can be used in this case. As well, sum of results from single module PSA level 3 is also can be used.

3.2.Scaling approach to EPZ-sizing

3.2.1.Overview of scaling approaches and State-of-the-art practices

References to the scaling approach in EPZ-sizing are often found in issues of low-power NPPs. In some of works, this approach considered as a reliable techniques in EPZ-sizing. NuScale's Approach by Doyle, J. (2022) emphasizes scaling used as a conceptual argument: the EPZ size should be proportionate to plant risk and design. NuScale proposes a risk-informed, performance-based approach to EPZ sizing, allowing for the EPZ to be adjusted based on the specific safety features and risk profiles of their reactors. This approach is fundamentally different from the traditional, one-size-fits-all model for EPZs, which typically assumes a fixed distance based on the reactor type and size. Instead, NuScale's method involves scaling the EPZ to the reactor's design and the actual potential offsite radiological consequences, ensuring that the size of the EPZ is directly tied to the reactor's unique characteristics. The goal is to ensure that the EPZ is neither excessively large nor unnecessarily small but instead aligned with the actual risk of an accident, based on deterministic analyses of accident scenarios and dose projections.
The article by Oh, K., S., Kim, S., et al. (2019) emphasizes the importance of scaling in the context of multi-unit nuclear power plant sites, arguing that traditional single-unit risk assessment frameworks may significantly underestimate total site-level risk. Through a PSA Level 3, the study demonstrates that off-site risks for public do not increase linearly with the number of units due to shared vulnerabilities and correlated hazards. It proposes a methodology to scale risk assessments by accounting for inter-unit dependencies and simultaneous accident progression. This approach enables a more realistic evaluation of emergency preparedness and public safety impacts for multi-unit configurations, suggesting that emergency planning and safety metrics must be re-scaled to reflect the compounded nature of multi-unit risks.
In the study on off-site radiological consequences from an SMR by Ilvonen, M. (2018), the concept of scaling is critically analyzed in the context of SMRs and their EPZs. The report addresses how the inherently smaller radioactive inventory and advanced

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passive safety features of SMRs may allow for a reduction in the size of the EPZs traditionally required for large nuclear power plants.
In the further VTT's report by Ilvonen, M. (2022), scaling is a central focus, particularly regarding the reduced size of EPZs that could be implemented for SMRs such as the EcoSMR design. The author discusses how, due to the inherent safety features of SMRs—specifically their smaller radioactive inventory, advanced containment, and passive safety systems—the potential radiological impact from accidents is significantly lower compared to traditional large reactors. This allows for a scaling down of EPZs, reducing the area within which emergency preparedness measures such as evacuation and sheltering are required. The report relies on detailed atmospheric dispersion simulations, including the use of the VALMA atmospheric dispersion model, to estimate the radiological consequences of potential release scenarios. These simulations are based on conservative assumptions, considering a variety of accident types, including those with potential radioactive releases. The simulations demonstrate that the radiological doses outside the reactor site for an SMR design such as EcoSMR would likely be much lower than those from larger nuclear reactors, thereby justifying a smaller EPZ.
Above mentioned statements can be applied to EPZ-sizing for multi-unit/multi-module context as well, because, the same value of thermal power can be divided among a several modules onsite. Regarding assumed linear relation between thermal power and inventory, scaling concept logically refers to the number of same-type modules onsite. The same value of thermal power can be divided among several modules onsite. Regarding the assumed linear relation between thermal power and inventory, the scaling concept logically refers to the number of same-type modules onsite. Although, having a certain list of accident source terms, such an approximation can only give some initial findings about the size of EPZs and should be justified considering the features of the particular SMR design and the actual regulatory framework. However, scaling cannot be recommended as a robust approximation, at least for two reasons: first, as recognized in this report, there is no linear relation between power and EPZ distance because atmospheric dispersion does not scale linearly with distance and may even aggravate at short ranges relevant for SMRs; second, scaling from a conventional plant disregards the essence of SMR technology, which lies in its advanced safety design and not in being a 'smaller version' of a conventional reactor. Therefore, a consistent approach is to follow the line supported by the SMR Regulators' Forum, in particular the EPZ working group, namely the adoption of a plant-specific risk-informed and performance-based approach to EPZ sizing.

3.2.2. Preliminary assessments of the impact of multimodularity effects on EPZs

In order to preliminarily demonstrate the impact of the accident release magnitude on the size of EPZs, a simplified calculation based on Gaussian plume dispersion model, single-point meteorology and a set of scaled hypothetical source terms can be used. For the case study 180-MWth iPWR reactor is considered with different numbers of module in frame of one reactor island. There are 1, 2, 4, 6 and 8 modules are installed on a single site. Using simplified assumption, the character in changing size of the EPZ depending on the number of modules can be observed.

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To assess the impact of SMR multimodularity on EPZs, the *HotSpot Health Physics Codes* software, version 3.1.2, was used. This analytical tool was developed at the Lawrence Livermore National Laboratory (USA) for rapid assessment of radiological consequences during accidents involving the release of radioactive materials into the environment. HotSpot software uses a Gaussian plume model for radionuclide dispersion and allows for the calculation of the total effective dose equivalent (TEDE), taking into account inhalation intake, direct exposure from the cloud, external exposure from contaminated surfaces, and resuspension of particles into the air.

In this study, one of HotSpot's basic calculation dispersion models – *General Plume*, covering the full range of radionuclide intake pathways – was applied. The assessment criterion was selected in accordance with NRBU-97 (Ukrainian Radiation Safety Standards): an effective dose within the first two weeks after the accident 50 mSv (the level of unconditional justification for evacuation) applied for establishing monitoring zone around the facility.

A distinct feature of the approach is that each small modular reactor (SMR) module was treated as an individual source term. In the multi-module scenarios (2, 4, 6 and 8 modules), the activity of the release was proportionally increased doubled, quadrupled, and sextupled, respectively. This allowed for an assessment of how an increase in the total activity affects the size of the zones where regulatory dose criteria for the population are exceeded. HotSpot software automatically calculates the spatial distribution of the total effective dose equivalent and generates dose-distance tables. Based on these tables, the distance was determined at which the averted effective dose within the selected dose criteria.

The input data used to model the accidental release scenarios were unified across all calculations, except for the variable release activity, which was set proportionally to the number of SMR modules. A mononuclide source term (^{137}Cs) was selected, as it contributes significantly to the total effective dose. The single-module source term is selected as 1E13 Bq; for 2, 4, 6, and 8 modules, it was 2E13 Bq, 4E13 Bq, 6E13 Bq and 8E13 Bq, respectively. Thus, in each simulation, the source term represented the total activity of all identical modules at a single site.

Meteorological conditions were conservatively defined:

- atmospheric stability class was F, representing the most stable condition with minimal atmospheric mixing and maximum ground-level aerosol concentration.
- wind speed at 10 meters above ground was 2 m/s, corresponding to a light breeze.
- terrain was assumed flat, without considering obstacles or urban structures.
- effective release height was set to 0 meters.
- duration of the release was defined as 10 minutes, simulating a short-term, instantaneous release characteristic of the initial stage of an accident.
- breathing rate was set to the default value of 3.33E-4 m³/s, corresponding to the average air intake of a resting adult.
- receptor height (i.e., the height at which the representative person from public is located) is 1.5 meters, representing the breathing zone of an adult.

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The simulation calculated TEDE values for four scenarios corresponding to the release of ^{137}Cs from one, two, four, six and eight SMR modules. The output data from the HotSpot calculations are presented in Appendix 6.2. The results clearly show a consistent decrease in TEDE with increasing distance. At the same time, an increase in the number of modules leads to higher absolute dose values at the same distances (Table 4).

Table 4. Effect of the release magnitude (number of modules under the common reactor building)

Case	Total activity released into the atmosphere (^{137}Cs), Bq	Distance of the dose criteria exceeding (50mSv), km
1-module	1.00E13	0.49
2-module	2.00E13	0.69
4-module	4.00E13	0.98
6-module	6.00E13	1.20
8-module	8.00E13	1.40

As the number of modules at the site increases, the total magnitude of the release also rises accordingly, leading to an expansion of the zone in which the dose criterion is exceeded. Thus, multimodularity significantly affects the size of planning zones and must be taken into account during the design of protective measures and risk assessment. The calculation results confirmed that increasing the number of SMR modules can potentially affect an EPZ size. A set of similar assessments has been provided for 1-2-4-6-8-module cases; distance values have been analyzed, from which a linear dependency has been observed pointing that the scaling approach can potentially be applied in the frame of the same types of modules and a common reactor building. While this outcome may appear self-evident (i.e. a larger source term leads to a larger EPZ), the more non-trivial finding is that the relation is not strictly linear when scaling across different reactor types. For example, if one starts with a conventional plant characterized by a source term of 8E13 Bq and applies a simple scaling approach to an SMR with eight times lower power, the expected EPZ distance by linear scaling would be ~0.175 km. However, the actual calculated value is ~0.49 km, i.e. 2.8 times larger than the linear estimate. This clearly demonstrates the effect of penalizing atmospheric dispersion at short distances, confirming that scaling from conventional plants to SMRs is not an acceptable approximation.

The form and dimensions of reactor building plays significant role in near range modelling. Based on known lateral width and height of the reactor building, parametrization of initial standard deviations of concentration should be considered (Figure 9).

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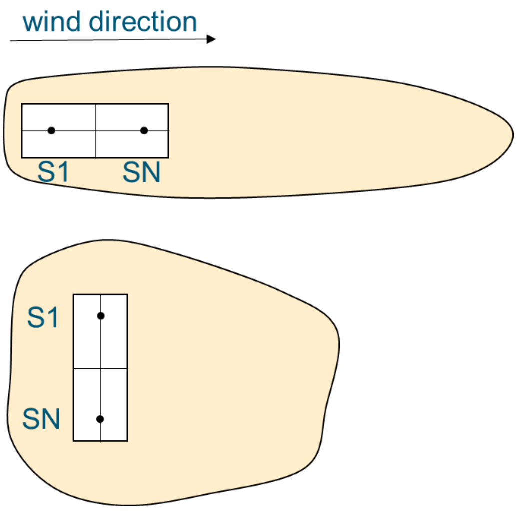

Figure 9. Building downwash (wake) effect for longitudinal and lateral configurations (N release sources from S1 to SN under the common reactor building)

The effect is relevant for near range modeling using CFD, Gaussian plume/puff, as well as Lagrangian particle models. However, with distance the effect becomes weaker. Numerical demonstration of building downwash effect (for reactor building with height 40m and width 100m), an overview of a short range dispersion approaches & phenomena, analysis of influence of thermodynamic parameters, release pathways, initial dispersion parameters on a short range results have been studied in frame of EU SASPAM-SA project (2024) WP6 "Characterization of iPWR EPZ". Preliminary calculations showed that plume rise can lay in range from 0 up to 35 m. Calculations of effective release height for the spectrum of heat emission as well as the preliminary HotSpot calculations for ground release for two options (with and without building downwash), for effective heights 15 m and 30 m have been presented under EU SASPAM-SA project (2024) WP6.

Regarding the separate spatial configuration of the release sources, cumulative effects may cause deviations in the EPZ boundary. Longitudinal configuration of the release sources (in comparison with wind direction) can increase the distances. Figure 10, Figure 11 and Table 5 and Table 6 demonstrate an almost neglectable influence of number of sources separated by 50-m distance between each other. However, depending on particular spatial configuration of the release points onsite, the differences can theoretically be more observable.

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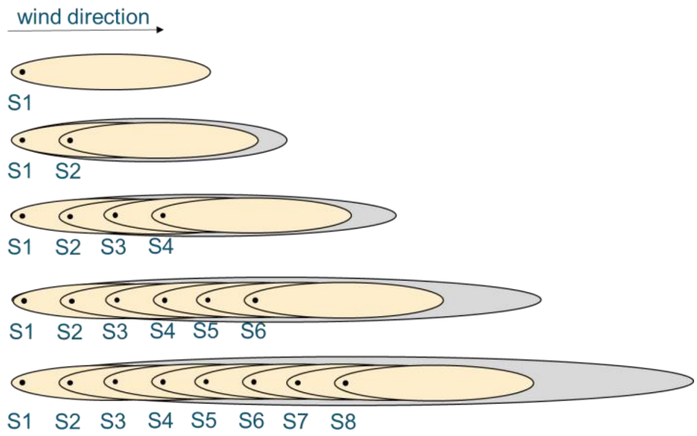

Figure 10. Longitudinal configuration of the separated release sources (50-m distance between of neighbouring release sources) in comparison with the wind direction and related possible cumulative effect

Table 5. Effect of longitudinal configuration of the release sources (50-m distance between of neighbouring release sources)

Case	Total activity released into the atmosphere (^{137}Cs), Bq	Distance of the dose criteria exceeding (50mSv) from S1, km	Distance of the dose criteria exceeding (50mSv) from the geometric center of the site, km
Single source	1.00E13	0.49	0.49
2 sources	2.00E13	0.73	0.71
4 sources	4.00E13	1.02	0.95
6 sources	6.00E13	1.35	1.23
8 sources	8.00E13	1.60	1.43

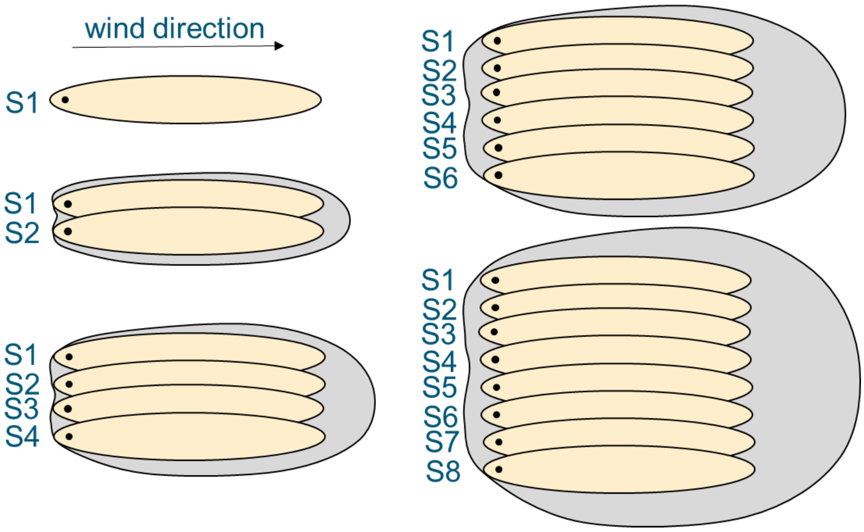

Figure 11. Lateral configuration of the separated release sources (50-m distance between of neighbouring release sources) in comparison with the wind direction and related possible cumulative effect

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Table 6. Effect of lateral configuration of the release sources (50-m distance between of neighbouring release sources)

Case	Total activity released into the atmosphere (^{137}Cs), Bq	Distance of the dose criteria exceeding (50mSv) from S1 or geometric center of the site, km
Single source	1.00E13	0.49
2 sources	2.00E13	0.70
4 sources	4.00E13	0.99
6 sources	6.00E13	1.21
8 sources	8.00E13	1.41

3.3.Deterministic approach to EPZ-sizing

Deterministic assessment produces 'yes/no' type of results. For example, a 'bounding' accident sequence (leading to the one certain 'maximum' atmospheric release that might possibly occur, according to the set of accident sequences included in the analysis) may be calculated up to offsite doses, and then it will become clear whether dose limits are exceeded or not. On the contrary, probabilistic analysis tries to produce the whole probability distribution (all potential consequences with their probabilities), to see e.g. the probability by which a certain offsite dose will be exceeded at a given distance & time point.

Regardless of deterministic or probabilistic way of work, both options have a variety of choices ranging from conservative assumptions to trying to produce as-good-as-possible or 'best estimate' predictions. This applies to the selection of release characteristics, release duration, atmospheric dispersion model and meteorological data, temporal and spatial resolution, assessment criteria, and other related factors impacting the selected calculation end-points. However, in terms of meteorological application, the single-point method commonly used in the deterministic approach is generally not resource-intensive, unlike the statistical methods used for off-site dose assessment.

3.3.1.Conservative and realistic approaches

Evaluating the likelihood of potential radiation emergencies is essential for ensuring a proportionate and effective approach to emergency planning. Although calculation of such metrics like frequency/probability is usually part of probabilistic approach, in deterministic safety analysis the frequency calculation can be provided for the selected branches of the event tree. Initiating events that lead to fault sequences protected by the same safety systems—and that result in similar radiological consequences — should be grouped together. It is relevant for the countries that use lower limits on the event frequency - DEC-B event(s) to be considered in EPZ-sizing. The frequencies of these sequences should be aggregated.

To represent each group, the source term selected should be the one that results in the most severe radiological dose (i.e., the bounding case).

The deterministic approach to defining EPZs includes an assessment of radiological consequences using specific selected release conditions and meteorological scenario(s).

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This assessment is conducted for a representative source term or group of source terms selected from the results of severe accident analysis.
In conservative radiological assessments, it can be generally assumed that most of the conservatism lies in the release fractions estimated for radionuclides released from the reactor core to the environment.
Table 7 shows the main set of assumptions on source geometry, duration of the release, Pasquill-Gifford stability class, dose criteria and exposure pathways used in deterministic approach. It contains an example of assumptions used in Ukrainian practice of deterministic EPZ-sizing in conservative assessment provided under EU SASPAM-SA project (2024) and demonstrates in radiological assessment for events associated with different frequency, we gravitate towards "more conservative" assumptions for AOO, DBA, and "more realistic" or “best estimate” assumptions for DEC-B/severe accidents.

Table 7. Assumptions on selection of input data for deterministic assessment

Event in radiological assessment	Calculation assumptions
NO/AOO	– Geometry: ground point release (conservative); – Duration of the release: 10 min. (conservative); – Pasquill-Gifford stability class – F, wind speed – 2 m/s, no precipitations (conservative); – Terrain: surface roughness z_0=3cm (conservative); – Criteria on effective dose – 40 μSv per annum due to all exposure pathways (NRBU-97);
DBA	– Geometry: ground point release (conservative); – Duration of the release: 10 min. (conservative); – Pasquill-Gifford stability class – F, wind speed – 2 m/s, no precipitations (conservative); – Terrain: surface roughness z_0=3cm (conservative); – Criteria on effective dose – 10 mSv in the first 2 weeks; thyroid equivalent dose 100 mSv in the first 2 weeks (due to inhalation); skin equivalent dose 300 mSv in the first 2 weeks;
DEC-B/ severe accident	– Geometry: building downwash (h=40m, b=100m); – Duration of the release: 10 min. (conservative); – Pasquill-Gifford stability class – D (likehood), wind speed – 3 m/s (stat.), no precipitations – Terrain: surface roughness z_0=100cm (city suburbs); – Criteria on - effective dose 50 mSv in the first 2 weeks; - thyroid equivalent dose (children) 50 mSv in the first 2 weeks (due to inhalation); - thyroid equivalent dose (adults) 200 mSv in the first 2 weeks (due to inhalation); - skin equivalent dose 500 mSv in the first 2 weeks

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In a multi-unit/multi-module context, the deterministic approach can be applied to a defined combination of source terms from several units or modules simultaneously. The deterministic approach typically uses a non-sophisticated atmospheric dispersion model, which assumes an instantaneous or short-term releases into the atmosphere.
According to the report by Kelk, R., Murad, A. (2020), in recent years, there has been a tendency to move from conservatism to best estimates, but include also uncertainty analysis, which at best gives quantification of uncertainties. This approach is called BEPU (best estimate plus uncertainties). It primarily applies to the models and assumptions used for accident analysis. However, for DiD Levels 4 and 5, it is recommended that best-estimate calculations be performed without uncertainty, including in dose assessment.
Deterministic techniques are generally conservative methods that tend to overestimate the consequences of a radioactive release. These methods offer a reasonable level of confidence that a bounding source term can be identified without the need for complex probabilistic calculations. 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 more accurate, yet less conservative, predictions.
For licensing purposes, the use of a conservative source term can be proposed — for example, to minimize research and development costs or to accelerate project timelines — provided that such an approach does not lead to design features or operational limits that could compromise safety. At the same time, for emergency preparedness assessments, source terms should reflect mean values derived from best-estimate calculations and should include quantified uncertainties to ensure a realistic and robust basis for planning.

3.3.2.Calculation end-points

Typically, the deterministic dose assessment is performed by applying a specific atmospheric dispersion model considering the site characteristics and meteorological conditions to the source term of specific accident sequences.
In accordance with TECDOC-2044 (2022) and a general concept of PSA level 3, consequences can be often in the form of prompt fatalities, long-term health effects or fatalities, economic losses. At the same time, the EPZ determination can be based on the following definitions of exposure pathways and the corresponding generic dose criteria, found in GSR Part 7 tables II.1 (rows 1–7, for precautionary urgent protective actions) and II.2 (rows 8–10, urgent protective actions and early protective actions).
In frame of international terminology, dose calculations should determine total effective dose (or TEDE). TEDE is the sum of the committed effective dose equivalent from inhalation and the deep dose equivalent from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.
According to the experience of Germany (Walter, H., Gering, F et al., 2016) and Sweden (2017:27e report, 2017), such quantity as cumulative frequency of protective action occurrences can serve as a probabilistic indicator of the potential necessity of implementation of urgent protective actions for the public living in the territory surrounding the facility. Statistical output can be used as an effective tool in the preparedness phase and emergency planning. In practice, it is also applied for such tasks

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as distribution of the additional response resources to protect the public, first detection of the releases and expected values of dose rate measurements, selection of location for installation of additional monitoring equipment, etc.
The concepts of averted dose and optimization of the level of intervention are useful in real conditions for a specific situation that develops during the course of an accident. However, these concepts are not practically applicable in determining emergency planning zones, since they contain a significant number of uncertainties related to specific conditions, from a specific time of day and active resources to human behavior during an emergency.
Therefore, in practice, instead of averted dose levels, the predicted dose and the selected level of intervention (in the conservative case, the lower limits of justification for a particular protective action) are more often used. For instance, in case of Ukrainian regulatory framework, calculation end-points represent all different protective actions such as evacuation, sheltering, relocation, iodine prophylaxis and food restrictions. For the basic assessments, Ukrainian intervention criteria can be applied. Intervention levels for the listed protective action are described in Section 2.4.1.

3.3.3.Inventory

The concept of “modularity” involves dividing the source term across multiple smaller, discrete reactor units. SMR core contains significantly less fuel than the core of a conventional large light water reactor. While modules can be added over time to achieve power output comparable to that of a traditional large reactor, their independent construction and operation reduce the likelihood of a large-scale offsite radiological consequence. This modular approach inherently limits the potential impact of any single initiating event.
Inventory is approximately proportional to the reactor's power level - for example, doubling the power level roughly doubles the core inventory. Lower power levels will result in lower fission product inventories than for typical large PWRs since the core inventory is roughly proportional to power.
In the event of an atmospheric release, only an accident-specific fraction of the core inventory is actually dispersed, with the fraction varying by radionuclide or group of elements due to differences in physical and chemical behaviour. The reactor core's radioactive inventory can be calculated using neutronic codes such as SERPENT (a Monte Carlo-based tool). According to Ilvonen, M. (2022), this type of analysis was commonly performed using the ORIGEN code.

3.3.4.Source term

Source term is 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.
In case of multi-unit/multi-module releases, source term should be presented in direct numbers of activity released from the affected unit/modules. It means that chemical release fractions cannot be applied in multi-unit/multi-module context due to possibly different inventories, delay before the release, effective release heights, etc.

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The temporal resolution of several source term should be sufficient to reflect important phenomena affecting release intensity. For multiple sources, it is recommended to select a single time reference system to enable the combination/aggregation of activity or any other characteristics from multiple modules/units into a single merged source term.

3.3.5.Meteorology

Establishing appropriate EPZ sizes for SMRs requires reliable meteorological data to estimate dispersion characteristics and population exposure under accident scenarios. Because SMRs generally have smaller source terms, detailed and localized meteorological assessments are essential to justify proportionally smaller EPZs.

In context of multi-unit/multi-module source term kinetics, some points on meteorology required for EPZ-sizing can be considered:

– hourly mast data for at least several years from the nearest hydro-meteorological stations (towers) around the facility; it should contain the necessary list of meteorological quantities impacting the results of dispersion modeling: wind speed and direction, atmospheric stability class, precipitations, etc.; weather mast measurements may be more reliable to acquire the dispersion parameters near the source than numerical weather model, but more masts than one single would make the data even more reliable and complete;

– numerical weather data (reanalysis data) with spatial and time resolution harmonized with spatial configuration of all release pathways onsite and kinetics of the source terms; spatial resolution of numerical weather data should cover near-range phenomena related to separated pathways of the release onsite;

– a format of meteorological data should be comparable to the requirements of actual tools for modeling atmospheric dispersion and dose assessment.

3.3.6.Offsite dose assessment

The optimization principle obliges the licensee to keep both individual and collective exposure of personnel and the public and the probability of critical events and associated potential doses as low as reasonably achievable, social and economic factors being taken into account.

Overview presented by Ilvonen (2022) highlight that offsite dispersion and dose assessment (public doses) calculations should be done with a selected validated and trusted code or codes, e.g. the following ones readily available for use:

– ARANO (VTT): Gaussian dispersion with internal & external doses, countermeasures;

– VALMA (VTT): Dispersion based on 3D trajectories from NWP data;

– MACCS (NRC): MELCOR accident consequence code system;

– RASCAL (NRC): Radiological Assessment System for Consequence AnaLysis.

However, additional analysis of actual decision support systems and EPR assessment tools shows that the list can be supplemented such widely-used codes supporting both deterministic and probabilistic approach to offsite dose assessment, such as

- HotSpot (NARAC): HotSpot Health Physics codes;
- JRODOS (KIT): Java-based Real-time On-line DecisiOn Support system;
- ARGOS (PDC-ARGOS): Accident Reporting and Guidance Operational System.

Extended list of code used in offsite dose assessment is presented in Section 3.4.1.

The listed codes have a necessary number of models to provide offsite dose assessments including multi-unit/multi-module calculation cases. E.g. JRODOS system allows user to apply the option of multi-unit releases (up to 5 in version of JRODOS v.2019). But, ability to model the multi-unit/multi-module releases should be additionally verified for each particular code in use. If the code does not allow to provide such assessments, calculations may be made for single-unit or single-module case (considering an interference phenomena) and then the results can be combined/merged using additional software with post-processing (e.g. QGIS or ArcGIS).

3.3.7. Required scope of severe accident analysis outputs regarding the multi-modules aspects

Based on practical side of offsite assessments, some results of severe accident analysis code as initial data for environmental atmospheric source term required for EPZ-sizing can be recommended:

- data on expected annual airborne and liquid discharges from all possible release pathways onsite (if PAZ has a mixed sanitary-protective and precautionary functions); as a prototype, large water reactor isotope composition can be considered, however, list of dose relevant radionuclides should be justified for each particular case;
- fission product activity in the reactor core (Inventory) from the beginning to the end of the fuel campaign (BOC, EOC) for the most complete list of radionuclides (available in the neutron-physics code library);
- data on the activity of the primary coolant (data is widely used for some postulated events under deterministic safety analysis);
- events under deterministic safety analysis);
- merged source term containing kinetics of the release as a percentage (release fraction) of the total reactor core activity by a set of chemical classes expected to be released into the environment due to failure/loss of integrity of physical barriers – these can be provided for the postulated severe accident scenarios by integral severe accident codes;
- data on additional purification or filtering mechanisms (if not covered by severe accident code);
- expected iodine form distribution (among elemental, organic and aerosol forms of iodine) and size of aerosol particles (AMAD distribution);
- release pathways for the postulated severe accident scenarios (including physical height of the release), plume rise components (heat flux, section area, vertical velocity), dimensions of the reactor building(s).

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3.4.Probabilistic safety assessments level 3

PSA is a comprehensive and integrated assessment of the safety of a reactor facility. The safety assessment considers the probability, progression and consequences of equipment failures or transient conditions to derive numerical estimates that provide a consistent measure of the safety. There are three levels of PSA:

- PSA Level 1 identifies and quantifies the sequences of events that may lead to the loss of core structural integrity and massive fuel failures.
- PSA Level 2 evaluates the chronological progression of core damage sequences identified in Level 1 PSA, including a quantitative assessment of phenomena arising from severe damage to fuel. Level 2 PSA addresses the phenomenon of a core damage accident, the response of the containment to the expected loads, and the transport of radioactive material from the damaged fuel to the environment.
- PSA level 3 starts from the level 2 results. In addition to the aspects analysed within a Level 2 PSA, a full scope or Level 3 PSA also analyses the dispersion of radionuclides in the surrounding environment and potential environmental and health effects.

Level 1 PSA, Level 2 PSA and Level 3 PSA are sequential analyses, where the results of each assessment usually serve as a basis for the PSA at the next level. Interconnections between the PSA levels are illustrated in Figure 12.

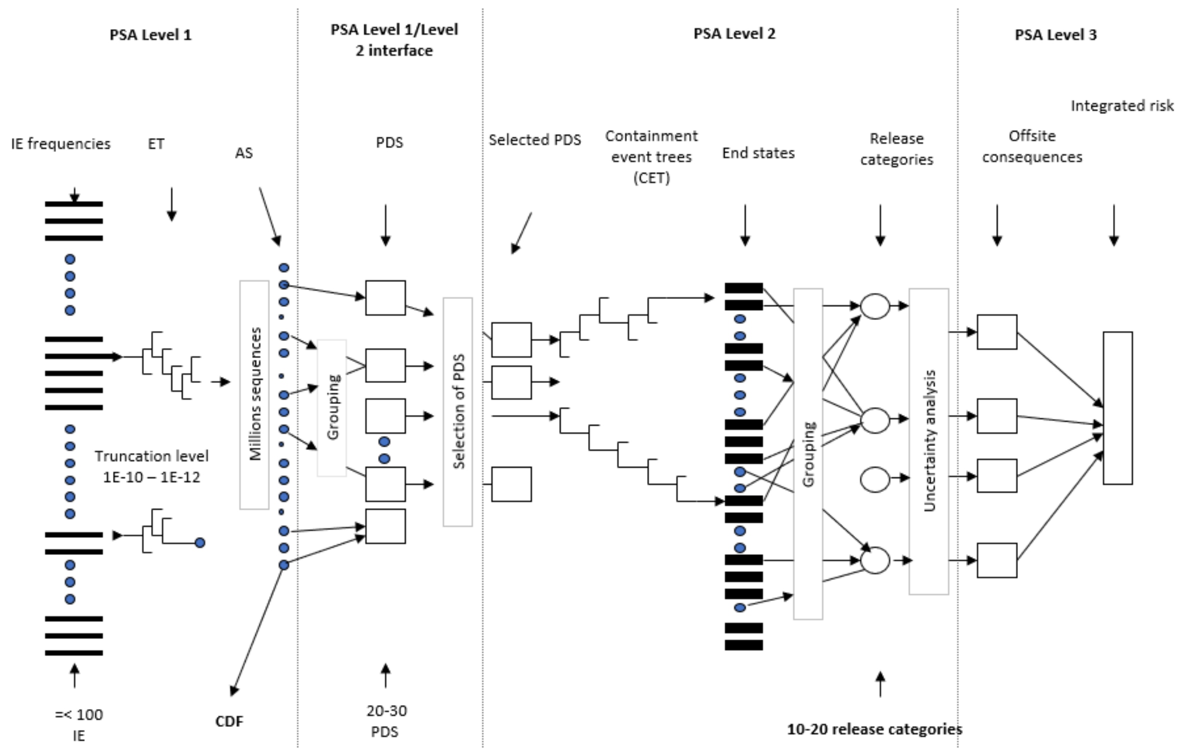

Figure 12. PSA overview

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In order to support definition of SMR EPZ, all three PSA Levels should be completed. The scope of analysis should be full-scope, and include the following:

– all plant operating states including full power, low power and shutdown modes. There may also be design-specific operating states unique to certain modular SMR designs which need to be addressed involving, for example, refueling and/or concurrent power operation and refueling.

– Whole spectrum of initiating events including internal initiating events, internal hazards, external hazards and combination of events.

– In addition to core damage, fuel handling accidents, and spent fuel pool accidents.

It should be noted, however, that using the PSA Level 3 results for definition of EPZ is more beneficial than post-processing of PSA Level 2 (as PSA Level 3 surrogate). PSA Level 1 results are not directly applicable for EPZ sizing.

3.4.1.Single module PSA

General process for Level 3 PSA is shown on Figure 13.

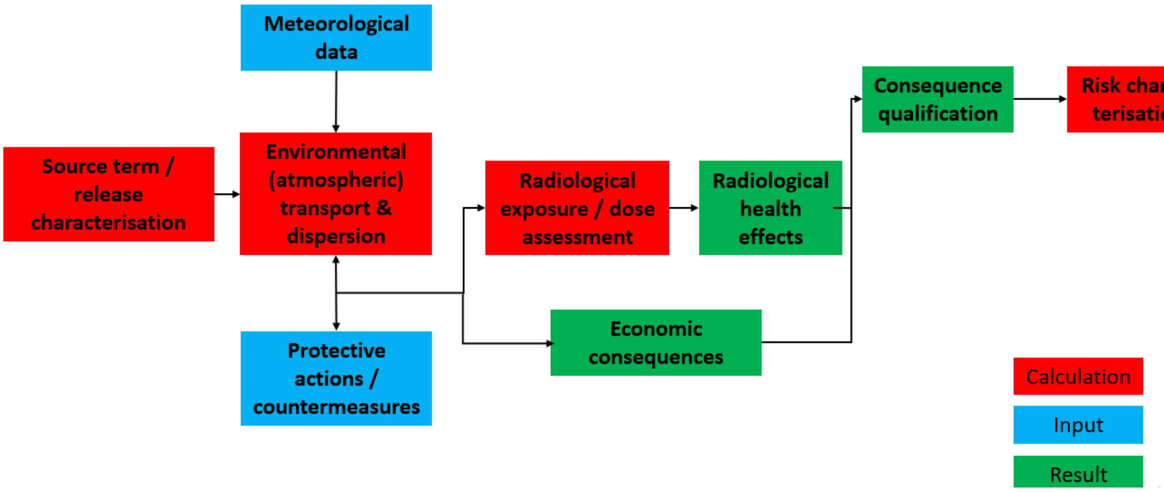

Figure 13. Level 3 PSA technical elements

Source term / radioactive release characterization is performed at Level 2 PSA. The results are represented as source term categories, where each category represents a group of possible accident sequences (end points of containment event trees). The PSA-specific characterization of source term categories includes:

– identification of radioactive sources (reactor, spent fuel storage, other),

– definition the timing of the release: start of release / delay time/ warning time / duration

– evaluation of the quantity and chemical form of radioactivity released:

  • magnitude of the release / radionuclide content

  • chemical form / particle size

– identification of location of the release (main ain stack, building surfaces, auxiliary buildings stacks):

  - atmospheric release / height of release / heat content
  - release to soil / water [not always considered]

- Quantification of frequency of occurrence, which is sum of containment event trees (or accident progression event trees) endpoints in the category. The quantification is important and related to the accuracy of the PSA models, which are built using various software codes. The PSA models include also assumptions and interface with results from deterministic analyses. The quantification of the frequency of the various sequences from the containment event trees uses the data on frequencies of the plant damage states, derived from the PSA level 1, and the conditional probabilities of the event trees. These probabilities include failure of safety systems such as the containment spray system (quantified also using fault trees) structural failures of the containment (quantified using a model of the performance of the structure), and the occurrence of physical phenomena where the split fractions relate to the analyst's evaluation. For the split fractions, the numerical values are derived from judgment supported by available sources of information.

Typical outcomes of Level 3 PSA includes the following information defined per source term: the contamination of the ground in the form of a statistical distribution due to the various weather conditions that may be encountered during the release; the dose an individual outside the facility may receive, in the form of a statistical distribution due to the various weather conditions that may be encountered during the release; the collective dose that a group of individuals may receive, in the form of a statistical distribution due to with the various weather conditions that may be encountered during the release; an increased probability, or severity, of health effects for an individual outside the facility, in the form of a statistical distribution due to the various weather conditions that may be encountered during the release; the risk incurred by an individual outside the facility if an accident occurs, expressed as the (location-specific) conditional individual risk (likelihood x consequences) and conditional group risk; the size of areas in which the intervention levels for implementation of protective measures is exceeded (such as sheltering, evacuation, decontamination, relocation, and food control). Finally, for the full source term spectrum, the risk incurred by an individual outside the facility, expressed as the (location-specific) individual risk and group risk, as well the effect of planned countermeasures on the risks should be defined as the result of Level 3 PSA.

An approach for using probabilistic approach for definition of EPZ is discussed in Serbanescu (2018), Serbanescu and Min (2021), where the PSA results are combined using probabilistic tools (convolution of integrals) and not summarization of average values, and therefore they provide a range of variation and a value for the epistemic uncertainty which more appropriate than the deterministic case. After obtaining frequencies for PDS, fatalities are calculated for each source term category. Serbanescu (2018) proposed to use post-processing of PSA Level 2 results with the fatalities, instead of performing PSA Level 3. The EPZ size is defined by equations:

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$$\mathrm{Rad}_d = S_d * R_d * C_d * \mathrm{Diff}_d * D_d + \Delta U_d$$

$$\mathrm{Rad}_p = S_p * R_p * C_p * \mathrm{Diff}_p * D_p + \Delta U_p \cong$$

$$\cong S_p * R_p * C_p * \mathrm{Diff}_p * D_p * \int f_1(S_p) * f_2(R_p) * f_3(C_p) * f_4(\mathrm{Diff}_p) * f_5(D_p)dx \pm \Delta U$$

$$\mathrm{Rad}_p = \mathrm{Rad}_d * f_1(S_p) * f_2(R_p) * f_3(C_p) * f_4(\mathrm{Diff}_p) * f_5(D_p) + \Delta U_p \quad ,$$

Where:

Sd - Source term in deterministic approach; Rd – Reactor failure criterion in deterministic approach; Cd – Containment failure criterion in deterministic approach; Diffd - Diffusion criterion in deterministic approach; Dd - Fatalities criterion in deterministic approach; Sp - Source term in probabilistic approach; Rp – Reactor failure criterion in probabilistic approach; Cp – Containment failure criterion in probabilistic approach; Diffp - Diffusion criterion in probabilistic approach; Dp - Fatalities criterion in probabilistic approach; Δ Ud,p -Uncertainties in deterministic, respectively probabilistic calculations; Δ U - Final total uncertainties; f1(Sp), f2(Rp), f3(Cp), f4(Diffp), f5 (Dp) - Distribution functions for the probabilistic criteria; f total - Convolution of functions f1 to f5.

There are several uncertainties that can influence on the definition of EPZ size. The sources of uncertainties, are associated with the following:

- Definition of PDSs;
- Number of nodes and endpoints defined in the containment event trees;
- Number of source terms and release categories defined;
- The assumptions resulted from the thermal-hydraulic codes runs;
- Severe accident uncertainties;
- Modeling assumptions.

The impact of uncertainties is illustrated Figure 14.

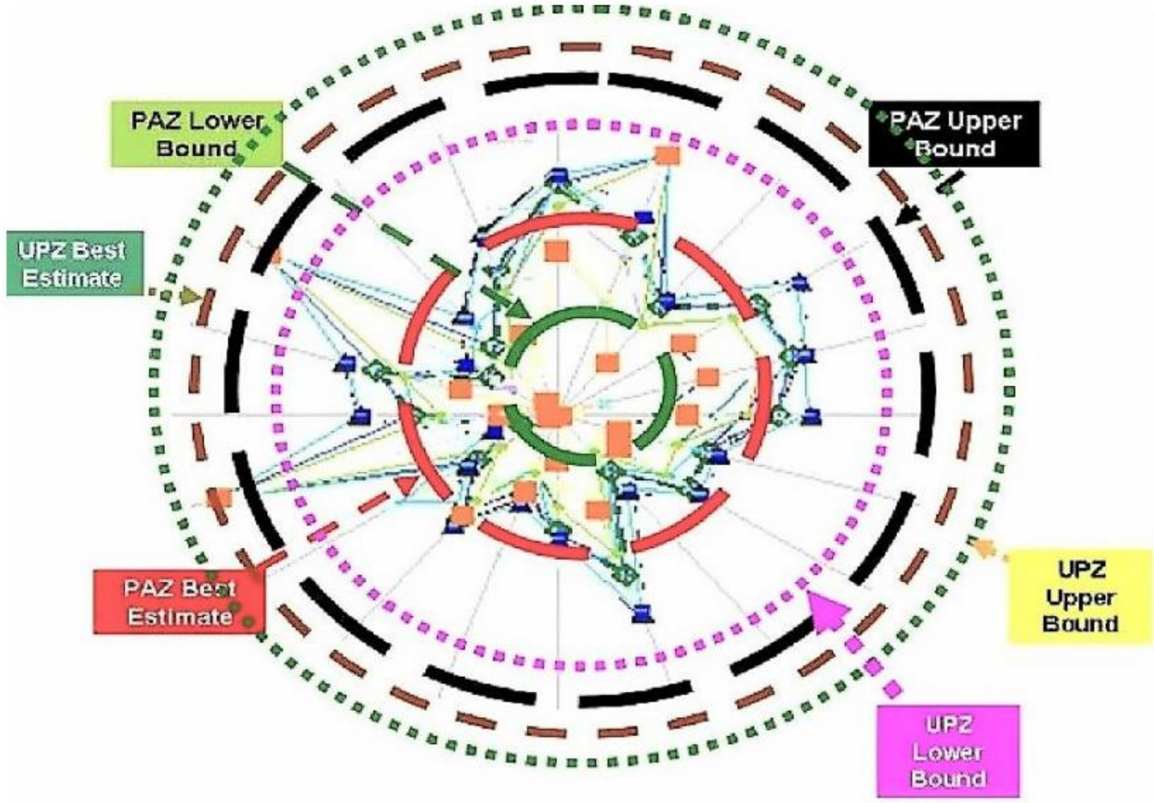

Figure 14. EPZ size depending on uncertainties and assumptions (optimistic/conservative), Serbanescu (2018)

Discussion on application of probabilistic, risk-informed approach for definition of EPZ is presented in Kubanyi et al (2008). It states that PSA application for EPZ includes incorporates modeled barriers and scenario-based aspects that are common across all types of nuclear safety analyses—whether deterministic or probabilistic. These include, for example, Design Basis Accidents (DBA), Beyond Design Basis Accidents (BDBA), Severe Accidents (SA), fission product characteristics, meteorological conditions, exposure pathways, adverse health effects, and strategies to prevent such effects. PSA evaluates risk metrics while integrating all these elements, leveraging its key strength: the ability to systematically and comprehensively address the full spectrum of initiating events within a unified analytical framework.

However, PSA also faces certain methodological and performance-related limitations that vary depending on the country and the specific group of users, like lack of a technical or legal framework to perform Level 3 PSA; large uncertainties in Level 3 PSA results, especially when combined with uncertainties propagated from Level 1 and Level 2 analyses; additional resources required to perform detailed off-site radiological consequence analyses; . These variations can lead to additional challenges in interpreting PSA results for applications such as EPZ definition and planning. The main shortcomings of PSA approach are summarized also in ELSMOR (2021):

- Level 2 PSA applications require a mature state of the art concerning the necessary deterministic safety analysis to simulate the accident progression until the released source term accounting for technology-specific features.

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- Level 1 and Level 2 PSA accident sequence selection needs the application of a screening-out criteria based on frequency threshold (e.g., 1E-07 for initiating events; 1E-09 for accident sequences, 1E-12 for minimal cutsets truncation). This criterion goes against current trends of preventing the exclusion of sequences whenever solely grounded on low frequencies.

To cope with this, using Best Estimate Plus Uncertainty (BEPU) approaches is proposed, see ELSMOR (2021). If a comprehensive BEPU approach were applied to the severe accident domain, the confidence level of the source term release categories would reveal potential deficiencies in achieving solid results and hence would support the decision on neglecting low-frequency events. There are, however, some scepticism regarding BEPU: the state of the art regarding the field of severe accidents is not mature enough to apply BEPU. There is no BEPU application to severe accidents, because the experiments are not enough to allow it. BEPU is about the models used in simulating the challenging scenario evolution. But this does not say anything about following a deterministic or probabilistic approach, about which there is no consensus at all.

3.4.2.Multi-module PSA

For MM SMR sites the following requirements from IAEA SSR-2/1 (2016) are acceptable: safety systems shall not be shared between multiple units unless this contributes to enhanced safety. Each unit of a MM NPP shall have its own safety systems and shall have its own safety features for design extension conditions. For further safety enhancement, means allowing interconnections between units of a MU-NPP shall be considered in the design. Although high level of independence between SMR modules is required, there are several shared aspects that can influence of the results of PSA Level 1, Level 2 and Level 3:

- Shared systems/structures and components (SSC) including cross-ties that could be used to connect systems between modules (external power, pumping station, common buildings, support systems;
- Shared resources (e.g. water, fuel);
- Common site mitigation provisions (the site swing diesel, which can be used to support different units, and the capacity to support single or multiple modules)
- Potential inter-module common cause failures (e.g. due to identical components, identical maintenance);
- Potential hazard correlations (e.g. seismic hazards, tsunami);
- Proximity dependencies, including shared main control room (MCR);
- Human and organizational dependencies (e.g. shared MCRs, sharing and limitations of human resources, availability or lack of accident management procedures that can support a MM accident);
- Possibility of accident propagation between units;
- Initiating events occurring in the site context and MM interactions (impact from one unit to another, e.g. due to radioactive release).

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These aspects should be identified and considered in MM (or multi-unit, MU) PSA. Level 2/3 MM PSA are required to quantify impact of the increased source terms and timing considerations from a multiple modules release on early health effects, need a Level 2 MUPSA.

General process for MM PSA is shown on Figure 15.

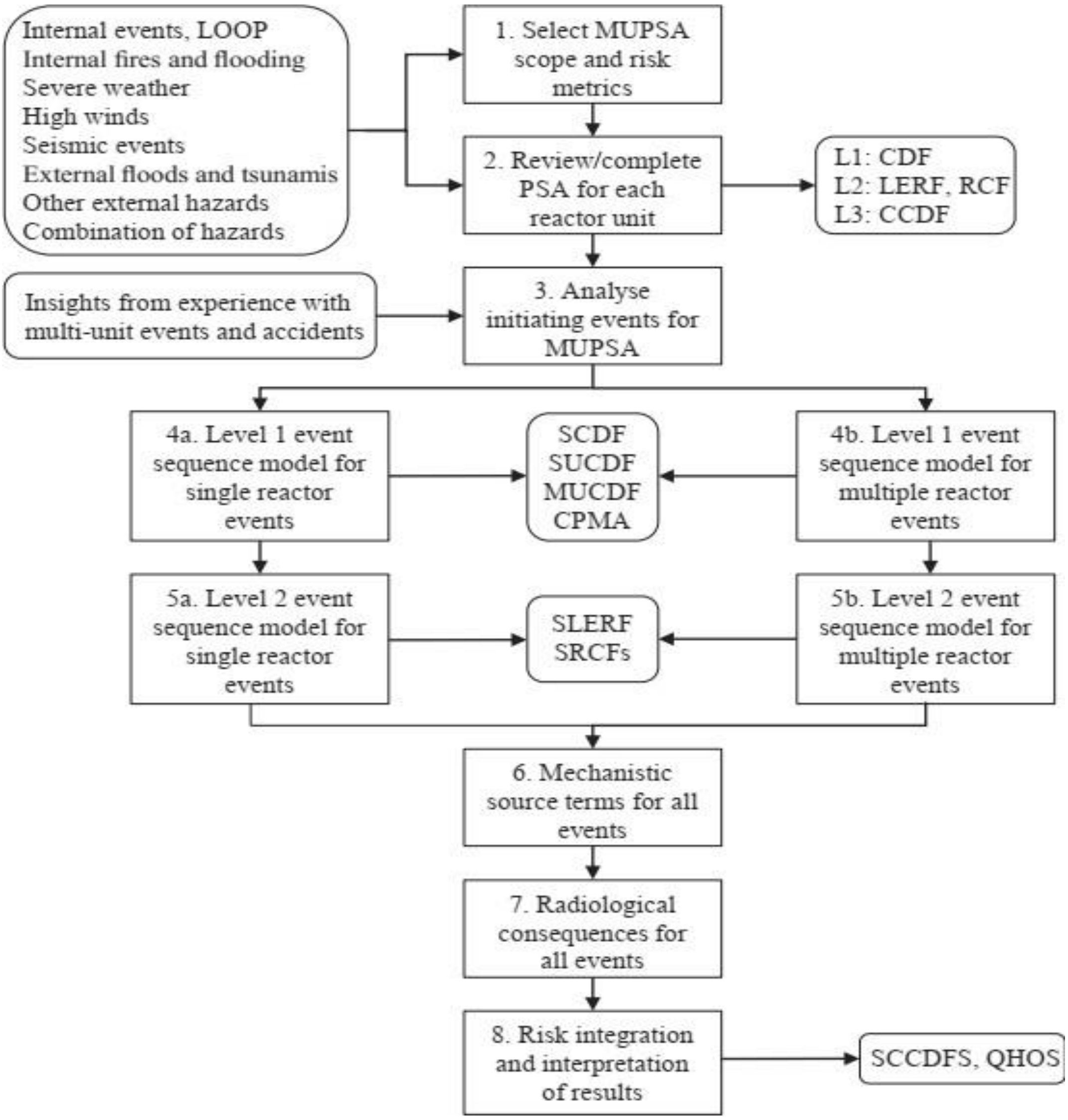

Figure 15. Multi-unit PSA steps, IAEA SRS 96 (2019).

The latest developments of MU PSA methods presented in IAEA IAEA SRS 110 (2023), IAEA-TECDOC-2044 (2022). Analysis of the key technical issues (risk metrics, initiating events, common cause failures, plant operational states, human reliability, accident sequences), associated with the implementation of mainly Level 1 PSA for MM SMR concept is presented in ELSMOR (2022). The main outcomes are as follows:

– the impact on the site of an initiating event occurring on one unit on the safety of neighbouring unit is possible but it is deemed lower than the impact of initiating events occurring concomitantly on the units (e.g. Loss Of Offsite Power). Moreover, adequate design provisions could allow to decrease this impact. It could be assumed that "MM initiating events" will dominate the site risk, and that contribution of “single unit initiating events” to the MM Core Damage Frequency could be neglected in a first approach;

– common cause failures between the units are expected to be one of the major contributors to MM risks;

– the probability of human error during post accidental phase is likely to increase due to the global management of several units.

The assessment of source terms, and the resulting release categories assignments, is fundamentally unchanged for the MMPSA. The existing release categories grouping used by the single module PSA can be used for the MMPSA. The release categories provide information on the timing, size and location of the release, which can be useful in the assessment of the impact of core damage from one unit affecting an adjacent unit or other unit on-site. For Level 3 PSA, it may be useful to develop simplified groupings for MM release categories. The initial output from the Level 2 MMPSA would involve sequences where Unit 1 experiences one plant damage state and release category, while Unit 2 experiences another plant damage state and release category. Example of MMPSA modeling using event tree linked approach is shown on Figure 16

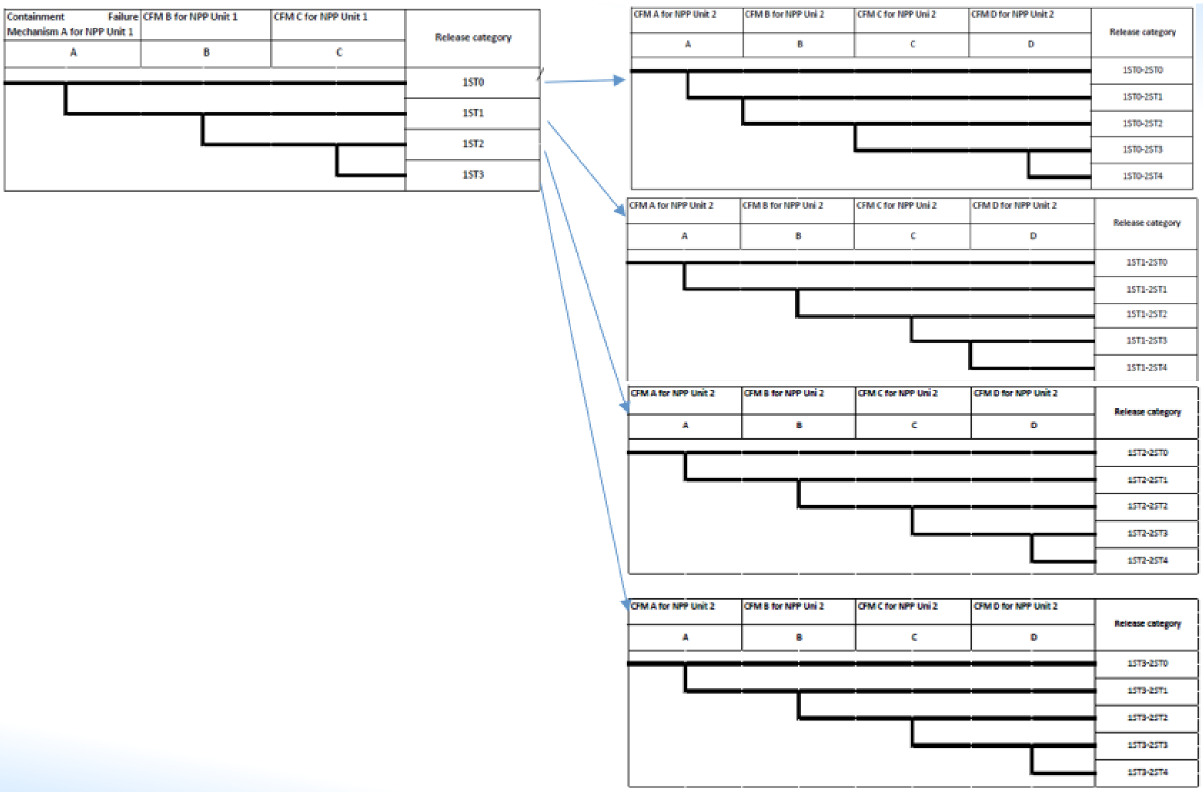

Figure 16. Linked event tree approach for MMPSA

Event tree linking approach (Master Event Tree approach) model the plant response (technological equipment, operator actions, etc.) for each initiating event in a single combined event tree for all modules on the site. Other modeling approaches are:

– Single fault tree approach, deals with event tree conversion to fault trees and combines risk logic for all modules, all hazards and all modes under a single-top

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logic gate. It is more easily to model combination of sources, including consideration for multiple plant operational states for each module;

– Hybrid approach - combination of master event tree and single fault tree approaches. Master event tree is developed with a single event tree node for each of the modelled modules or sources. The node for each unit or source is then modelled with a unit specific top event or SFT. It is possible to easily develop the specific outcome from specific combinations of modules. Weakness of the approach is a need to develop separate event trees for each IE;

– Minimal cut set conjunction. Simplified surrogate approach that does not take into account the whole spectrum of dependencies between the modules.

Release timing can affect the categorization of release categories as well the impact on the human reliability assessment (dose impact on the adjacent units).

As an example, assume that a two-unit site has 30 RCs for the SUPSA. A simplification is needed, since in theory, the 30 RCs in this example could end up with some 900 possible combinations for Unit 1 and Unit 2 RCs. Without simplification, the Level 3 analysis could be difficult to manage.

RCs for either shutdown POS or SFP core damage sequences could also complicate the MUPSA analysis. However, the general approach is similar, whether considering the impact of core damage on an adjacent unit or solving the Level 2 model for the resulting RC combinations.

Multiunit module and single unit module releases are physically similar in terms of source term category. However, for highly independent SMR modules, multi module risks contribute generally less than 5% to the total risk with respect to frequency, see example on Figure 17. Multiunit large release frequency is lesser than 1% of total large release frequency for sites with low inter–unit dependencies, truncation value is to be significantly decreased, comparing to single unit PSA.

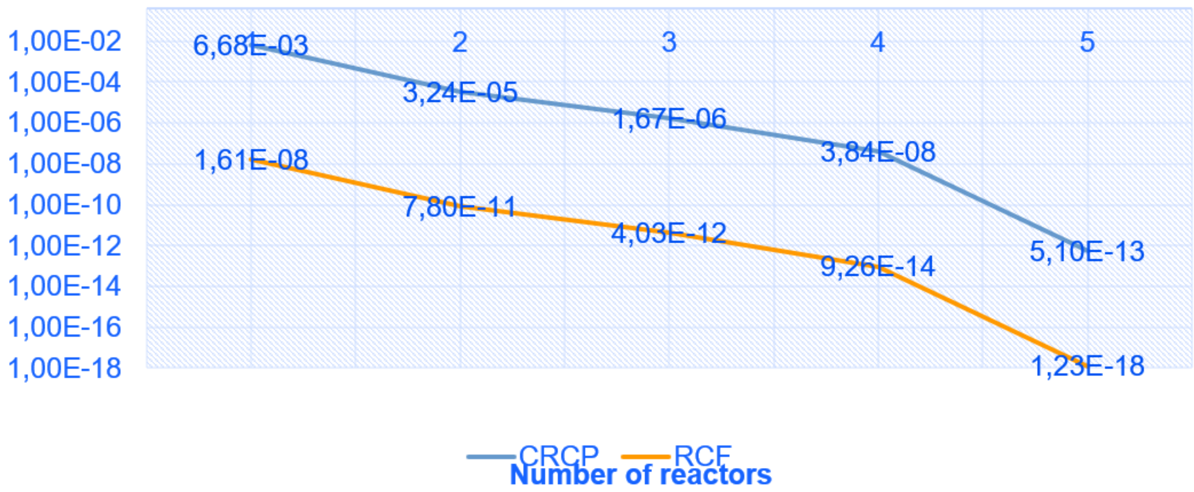

Figure 17. Multi unit release category frequencies depending on number of units

Shared systems provide some additional redundancy for each unit at the site; however, it may be revealed that not all systems or human resources will be sufficient to mitigate an accident after a multiunit accident initiator. Multiunit risk assessment needs to be carried out in an integrated manner rather than evaluating each unit separately.

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The advantage of performing a Level 3 MMPSA is that site risk metrics such as individual dose, individual risk or societal risk provide a direct measure of the risk to people and the environment and can be compared with any site public risk goals. Level 3 MUPSA involves calculation of the total risk from all units at the site. Some issues to consider when defining risk metrics or performing calculations for individual risk considerations:

– the representative person — the individual for whom the risk from the site is calculated — may need careful selection, taking account of the locations of the individual units with respect to potential exposed individuals. A simpler approach may be to assume that all releases occur from a single point, but this may lead to an overestimate of the risk;
– decisions on pathways to include, whether and which countermeasures (protective actions and/or remedial actions) can be credited and integration times for the deposited dose will also need to be made.

Issues for societal risk considerations are as follows:

– aggregating low doses over large numbers of people could lead to conservative results with large numbers of notional fatalities if the linear no threshold assumption is applied;
– certain judgements will need to be made on the geographical and temporal extent over which to perform the calculations;
– consideration of the demographic situation and the tendency of population density to increase. Also, simultaneous releases may give different off-site consequences in terms of the number of early fatalities to those from releases offset in time; this is a result of changes in meteorological conditions — principally wind direction — during the releases exposing different numbers of people at different levels and the non- linear risk–dose curve for deterministic effects. In these cases, the results are likely to be very sensitive to the exact population distribution and changes in meteorological conditions.

It should be noted that the MM site risk metric is not just risk from multi-modules accidents. The complete site metric (source term, release category frequency, large release frequency, fuel damage frequency) should include the following, example for LFR:

- LRFs from each single module at the site; plus
- LRF from MM accidents at the site.

As discussed in SMR Regulators' Forum (2019), the size of EPZ may be impacted by the number of reactor modules/units, therefore the proper consideration should be given to the operating state (e.g. operation/shutdown/maintenance) of all unaffected units on the site and the limitations of non-standard equipment (e.g. cross-ties of electric or heat removal systems) that might be shared between the units. Example of EPZ for MM sites is show on Figure 18.

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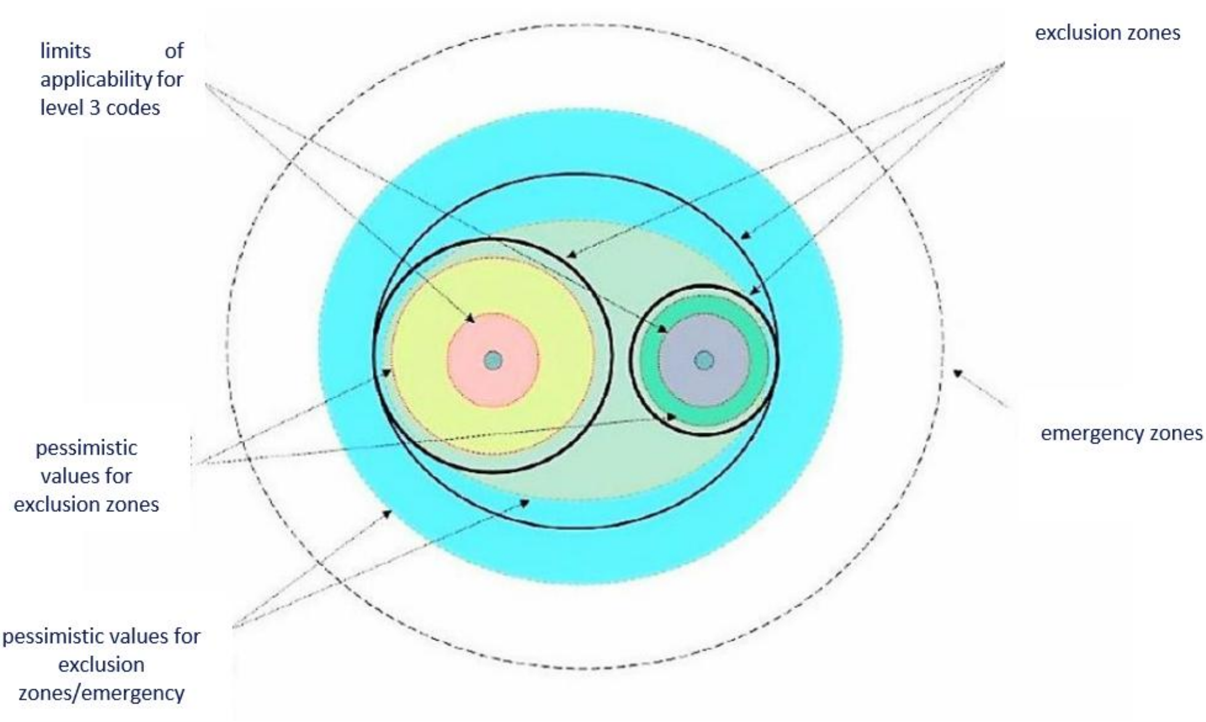

Figure 18. EPZ calculated based on PSA for a MM plant site, Serbanescu (2018)

3.5.Computational resources

Computational resources involved in EPZ-sizing depend on approach selected and capabilities of particular organization. However, any simplified attempts to provide calculation can be justified using conservative assumptions with e.g. simple Gaussian plume modeling. In order to reflect statistical distributions and provide more realistic assessments, probabilistic approach to atmospheric dispersion modeling can be applied using set of actual codes.

MACCS (MELCOR Accident Consequence Code System) calculates the offsite health consequences of an airborne release of radioactive material using site-specific information for the area and radiological release data. MACCS is the code used by the NRC to support Level-3 PRAs and based on hourly weather data (single-point measurements) from the weather towers onsite or near the site.

JRODOS (Java-based Real-time Online Decision Support system) is a decision support system developed by the Karlsruhe Institute of Technology (KIT) and European partners for nuclear and radiological emergency response. It is used to model the source term and assess the radiological consequences of accidental releases. JRodos integrates an atmospheric dispersion models (Gaussian puff RIMPUFF, Lagrangian particle DIPCOT, LASAT), hydrological transport models, dose assessment modules, and decision support tools for countermeasure evaluation.

ORCA (XDOSE) (On/Off-site Radiological Consequences of Accidents) calculates atmospheric dispersion and radiological doses following accidental airborne releases from nuclear facilities, following CSA Standard N288.2. The code supports calculations for facilities close to site boundaries with the implementation of the dispersion

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parametrizations in NUREG/CR-6331, which is consistent with software accepted by the U.S. NRC. By the Kinectrics (Canada).
ARGOS (Accident Reporting and Guidance Operational System) is a decision support tool developed by the Danish Emergency Management Agency (DEMA) for managing radiological, nuclear, chemical, and biological emergencies. The system integrates the RIMPUFF atmospheric dispersion. The system enables the estimation of plume spread, contamination zones, and protective action distances based on radiological dose thresholds. model, real-time monitoring data, dose calculation, and GIS-based visualization.
RASCAL (Radiological Assessment System for Consequence Analysis) is a software tool developed by the US NRC to provide source term estimation and radiological dose projection during nuclear incidents. RASCAL is intended for use by NRC personnel and licensees in the early phase of emergencies. The code includes modules for source term modeling, atmospheric transport and dispersion, and projected dose calculation. RASCAL uses a straight-line Gaussian plume atmospheric dispersion model with options for ground-level and elevated releases.
HotSpot is a software package developed by the Lawrence Livermore National Laboratory for radiological and nuclear emergency planning and response. It provides technically defensible, conservative, and realistic estimates of radiation dose and contamination. HotSpot is based on Gaussian plume modeling for atmospheric dispersion. It includes modules for explosions, fires, resuspension, and routine releases. The software requires minimal input data and allow to obtain statistical distribution based on hourly mast weather data.
COSYMA (COde SYstem for Multi-component Assessment) is a probabilistic accident consequence assessment system developed by Forschungszentrum Karlsruhe and the UK National Radiological Protection Board with support from the European Commission. It is used to estimate radiological consequences of nuclear accidents involving atmospheric releases. The system incorporates models for source term definition, atmospheric dispersion (Gaussian), dry and wet deposition, environmental transfer, and dose calculation via inhalation, ingestion, and external exposure. It supports both deterministic and probabilistic simulations.
PC-CREAM is a radiological impact assessment code originally developed by the National Radiological Protection Board, based on a methodology commissioned by the European Commission. It is used to assess the consequences of routine and accidental releases of radioactive materials to the environment. The code includes modules for atmospheric dispersion, environmental transfer, and dose assessment. Its core module, PC-CREAM Assessor, uses default parameters selected to ensure that dose estimates are conservatively overestimated. The software calculates individual committed effective dose and collective effective dose equivalent for specified populations.
Some examples of calculation resources involved in statistical assessment under SASPAM-SA project has shown in EU SASPAM-SA project (2024).

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4.Conclusions

International experience and regulations on EPZ-sizing are presented. Particular attention was paid to multi-module SMR-specific considerations in EPZ-sizing such as selection dose-relevant radionuclides and representative source term, near-range effects, release pathways and configuration of the release points, merged source term, temporal resolution of source term, meteorology and auxiliary data and computational limitations. These factors can play a significant role in atmospheric dispersion modeling in multi-unit/multi-module context.

The main features of both deterministic and probabilistic approaches to determining the size of EPZs in light of multi-unit/multi-module facilities as well as actual experience of current international projects have been analysed. Deterministic approach forms the basis of existing Ukrainian requirements and standards, is conservative and relatively simple to use compared to probabilistic assessment methods. Set of required initial data for EPZ-sizing has been listed and can be applied for preparation to EPZ assessment at the next stage of multi-unit/multi-module EPZ-sizing.

Scaling approach is an example of direct simple method to consider thermal power of unit(s)/module(s). Also it covers an issue of their numbers onsite. A set of similar assessments has been provided for 1-2-4-6-8-module cases, demonstrating the effect of penalizing atmospheric dispersion at short distances, confirming that scaling from conventional plants to SMRs is not an acceptable approximation. Having a certain list of accident source terms, such a linear approximation can only give some initial findings about the size of the EPZs and should be justified considering the features of the particular SMR design and actual regulatory framework. Given the potentially slower release kinetics, it is important to highlight several inherent or possible factors: a larger heat transfer area per unit power due to the small core, substantial water inventories, spacious containment volumes that limit pressurization, and increased surfaces that enhance natural deposition and decontamination processes, along with additional delays and barriers.

Using Gaussian plume model of HotSpot code and hypothetical accident severe accident mononuclide source term (tracer ^{137}Cs), preliminary testing calculations have been provided to analyse the impact of multimodularity on the sizes of EPZs (1-2-4-6-8 modules). Obtained results show that in case of simultaneous release from all modules under the common reactor building, number of modules can significantly impact the sizes of EPZs. Obtained calculation results demonstrates that for a single-module release, the distance at which dose criteria were exceeded was approximately 0.49 km, for an 8-module simultaneous release, the distance extended almost to 3 times – 1.40 km. These results indicate that the number of active modules directly affects the EPZ size if simultaneous accidents are assumed.

Generally, longitudinal as well as lateral configuration of the separated release sources in comparison with the wind direction potentially can have observable cumulative effect. For the considered testing case (50-m distance between of neighbouring release sources), the configuration has non-significant effect to distance of the dose criteria exceeding. A deterministic approach to atmospheric dispersion modeling allows for the configuration of sources at the site to be taken into account, while modeling with existing probabilistic tools may encounter difficulties in modeling effects associated with such effects.

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The size of EPZ depends on site-specific factors such as source terms, meteorological conditions, topography, and the planned protective actions implemented during accidental radioactive releases at varying distances from the source. Therefore, it is strongly recommended using plant-specific methods that calculate dose consequences as a function of distance for determining the EPZ distance. Key inputs for this determination include plant design, site meteorology, the assumed initiating event, safety analysis, and projected offsite dose consequences. As a standard, the state-of-the-art PSA is full-scope and plant-specific. Plant-specific approach uses plant-specific data, i.e. accident progression, source term, and site-specific meteorological conditions to define EPZ sizes. This, as well, another strength of PSA, like possibility to combine all logical possible cases, consideration of the combination of the distribution of impact to SSC and the fragility of the SSCs, stipulates using the probabilistic approach as state-of-the-art method for EPZ. However, high-technical quality, plant-specific and full-scope PSAs are not available for such SMRs designs that are considered in EASI-SMR project. That is why PSA-based approach will not be applied in detail in the project.

Under next steps of the task, calculations of EPZ for preselected MM SMR sites in Ukraine and Poland will be performed. Specifically, industrial location (Heat power plant 5) within the Kyiv city boundaries with power transmission infrastructure is preliminary proposed, see Figure 19.

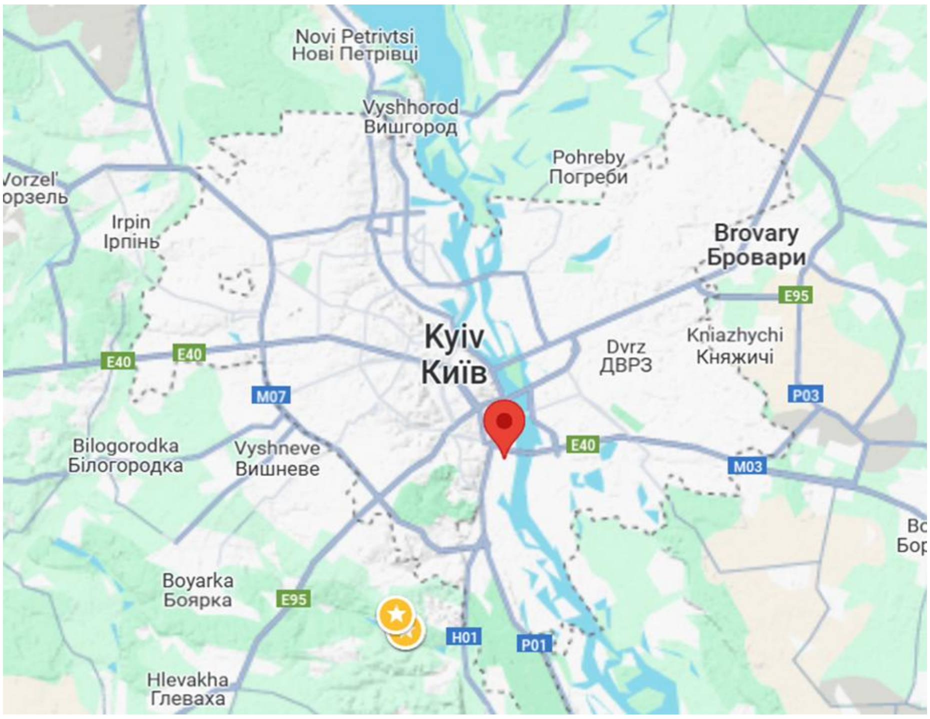

Figure 19. Proposed location in Ukraine

The selection was performed taking into account that population density similar to possible SMR locations; short human assistance response means and times and good transportation routes.

The next steps for T1.4, in particular justification of selected methodological approach can depend on the inputs received from other EASI-SMR work packages. Basically, it is expected to:

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- provide the characteristic of E-SMR and selected location of the site;
- cover airborne discharges during normal operation (NO), expected releases during AOO, DBA and DEC-B (severe accident); and justify their selection;
- use both deterministic and probabilistic approaches to atmospheric dispersion modelling and describe the software to be applied, including results post-processing;
- describe dose limits and intervention levels applied;
- select both conservative meteorology and numerical weather data set (for SA) to be used in dispersion modeling;
- establish criteria for relative number of considered meteorological cases (weather percentile P80/P90/P95 etc, that will depend on probabilistic characteristics of the selected severe accident and residual risk for public) - in case of severe accident;
- define the sizes of PAZ and UPZ depending on numbers of SMR modules (1-2-4-6-8).
- provide the discussion and recommendation on applicability of the used methodology to multi-module events at SMRs

Based on analysis of multi-unit/multi-module considerations for EPZ-sizing and on practical side of deterministic offsite radiological assessments, some results list should be provided by severe accident analysis code as initial data for source term required for EPZ-sizing have been recommended. Although in the absence of input data, some conservative assumptions can be made based on the specific models used, as well as the available information presented in open sources.

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6. Appendices

6.1. Glossary of Terms

Large release - an accident release of radioactive substances in the event of an accident, under which conditions requiring evacuation of the population in accordance with the levels of unconditional justification are created on the border of the sanitary protection zone of a nuclear power plant in accordance with radiation safety standards;

Emergency planning zones - territories around a nuclear power plant for which urgent countermeasures and other response measures are envisaged in accordance with radiation safety standards;

Monitoring zone - the territory where radioactive discharges and releases from nuclear power plants may be affected and where radiation monitoring is carried out;

Early radioactive release - a release of radioactive substances in the event of an accident that requires urgent countermeasures to be implemented off-site, which are not sufficiently time-consuming;

Design extension conditions (DEC) - conditions caused by initial events not considered as part of a design basis accident, in particular, the expected probability of occurrence of which is lower than that taken into account for design basis accidents, or the course (development) of which is accompanied by additional failures of safety systems or human errors compared to design basis accidents. Extended design conditions are divided into two categories: category A, which includes extended design conditions without severe damage to nuclear fuel, and category B, which includes accidents with severe damage to nuclear fuel (severe accidents);

Sanitary protection zone - the area around a nuclear power plant where the level of exposure of people under normal operation may exceed the dose limit quota for the population.

Emergency planning zones - areas around a nuclear power plant for which urgent countermeasures and other response measures are to be implemented in accordance with radiation safety standards

Conservative approach - an approach according to which the parameters and characteristics of nuclear power plant systems, elements and structures are based on values and limits that clearly lead to more unfavorable results

Design basis accident (DBA) - an accident for which the design of a nuclear power plant defines initial (initial) events and final states and provides for safety systems that ensure, taking into account the principle of a single failure of a safety system (system channel) or one additional personnel error independent of the initial (initial) event, limiting its consequences within the established limits

Large early release - release of radioactive substances in the event of an accident, which requires urgent countermeasures to be implemented outside the nuclear power plant site, but for which there is not enough time

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6.2.HotSpot calculation results

Description of the below presented calculation results

6.2.1.Single-module case

HotSpot Version 3.1.2 General Plume

Source Material : Cs-137 F 30.0y
Material-at-Risk (MAR) : 1.0000E+13 Bq
Damage Ratio (DR) : 1.00
Airborne Fraction (ARF) : 1.000
Respirable Fraction (RF) : 1.000
Leakpath Factor (LPF) : 1.000
Respirable Source Term : 1.00E+13 Bq
Non-respirable Source Term : 0.00E+00 Bq
Effective Release Height : 0.00 m
Wind Speed (h=10 m) : 2.00 m/s
Wind Speed (h=H-eff) : 0.83 m/s
Stability Class : F
Respirable Dep. Vel. : 0.30 cm/s
Non-respirable Dep. Vel. : 8.00 cm/s
Receptor Height : 1.5 m
Inversion Layer Height : None
Sample Time : 10.000 min
Breathing Rate : 3.33E-04 m3/sec
Distance Coordinates : All distances are on the Plume Centerline

Maximum Dose Distance : 0.010 km
Maximum TED : 1.21E+02 Sv
Inner Contour Dose : 0.250 Sv
Middle Contour Dose : **0.050 Sv**
Outer Contour Dose : 0.010 Sv
Exceeds Inner Dose Out To : 0.23 km
Exceeds Middle Dose Out To : **0.49 km**
Exceeds Outer Dose Out To : 1.1 km

FGR-13 Dose Conversion Data - Total Effective Dose (TED)

Include Plume Passage Inhalation and Submersion
Include Ground Shine (Weathering Correction Factor : None)
Include Resuspension (Resuspension Factor : Maxwell-Anspaugh)
Exposure Window:(Start: 0.00 days; Duration: 14.00 days) [100% stay time].
Initial Deposition and Dose Rate shown
Ground Roughness Correction Factor: 1.000

RESPIRABLE

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DISTANCE km	T E D (Sv)	TIME-INTEGRATED AIR CONCENTRATION (Bq-sec)/m3	GROUND SURFACE DEPOSITION (kBq/m2)	GROUND SHINE DOSE RATE (Sv/hr)	ARRIVAL TIME (hour:min)
0.030	1.1E+01	3.8E+10	1.6E+07	3.3E-02	<00:01
0.100	1.2E+00	2.6E+11	1.2E+06	2.4E-03	00:02
0.200	3.2E-01	8.2E+10	2.8E+05	5.5E-04	00:04
0.300	1.4E-01	3.7E+10	1.2E+05	2.4E-04	00:06
0.400	7.7E-02	2.1E+10	6.5E+04	1.3E-04	00:08
0.500	4.9E-02	1.3E+10	4.1E+04	8.1E-05	00:10
0.600	3.4E-02	9.3E+09	2.8E+04	5.6E-05	00:12
0.700	2.5E-02	6.8E+09	2.1E+04	4.1E-05	00:14
0.800	1.9E-02	5.2E+09	1.6E+04	3.1E-05	00:16
0.900	1.5E-02	4.2E+09	1.3E+04	2.5E-05	00:18
1.000	1.2E-02	3.4E+09	1.0E+04	2.0E-05	00:20
2.000	3.2E-03	9.0E+08	2.7E+03	5.3E-06	00:40
4.000	9.5E-04	2.6E+08	7.9E+02	1.6E-06	01:20
6.000	4.8E-04	1.3E+08	4.0E+02	7.9E-07	02:01
8.000	3.1E-04	8.5E+07	2.5E+02	5.0E-07	02:41
10.000	2.2E-04	6.1E+07	1.8E+02	3.6E-07	03:21

6.2.2.Two-module case

HotSpot Version 3.1.2 General Plume

Source Material : Cs-137 F 30.0y
Material-at-Risk (MAR) : 2.0000E+13 Bq
Damage Ratio (DR) : 1.00
Airborne Fraction (ARF) : 1.000
Respirable Fraction (RF) : 1.000
Leakpath Factor (LPF) : 1.000
Respirable Source Term : 2.00E+13 Bq
Non-respirable Source Term : 0.00E+00 Bq
Effective Release Height : 0.00 m
Wind Speed (h=10 m) : 2.00 m/s
Wind Speed (h=H-eff) : 0.83 m/s
Stability Class : F
Respirable Dep. Vel. : 0.30 cm/s
Non-respirable Dep. Vel. : 8.00 cm/s
Receptor Height : 1.5 m
Inversion Layer Height : None
Sample Time : 10.000 min
Breathing Rate : 3.33E-04 m3/sec
Distance Coordinates : All distances are on the Plume Centerline

Maximum Dose Distance : 0.010 km
Maximum TED : 2.42E+02 Sv

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Inner Contour Dose : 0.250 Sv
Middle Contour Dose : **0.050 Sv**
Outer Contour Dose : 0.010 Sv
Exceeds Inner Dose Out To : 0.32 km
Exceeds Middle Dose Out To : **0.69 km**
Exceeds Outer Dose Out To : 1.6 km

FGR-13 Dose Conversion Data - Total Effective Dose (TED)

Include Plume Passage Inhalation and Submersion
Include Ground Shine (Weathering Correction Factor : None)
Include Resuspension (Resuspension Factor : Maxwell-Anspaugh)
Exposure Window:(Start: 0.00 days; Duration: 14.00 days) [100% stay time].
Initial Deposition and Dose Rate shown
Ground Roughness Correction Factor: 1.000

DISTANCE km	T E D (Sv)	RESPIRABLE TIME-INTEGRATED AIR CONCENTRATION (Bq-sec)/m3	GROUND SURFACE DEPOSITION (kBq/m2)	GROUND SHINE DOSE RATE (Sv/hr)	ARRIVAL TIME (hour:min)
0.030	2.2E+01	7.6E+10	3.3E+07	6.5E-02	<00:01
0.100	2.5E+00	5.1E+11	2.4E+06	4.8E-03	00:02
0.200	6.4E-01	1.6E+11	5.6E+05	1.1E-03	00:04
0.300	2.8E-01	7.5E+10	2.4E+05	4.7E-04	00:06
0.400	1.5E-01	4.2E+10	1.3E+05	2.6E-04	00:08
0.500	9.8E-02	2.7E+10	8.2E+04	1.6E-04	00:10
0.600	6.7E-02	1.9E+10	5.6E+04	1.1E-04	00:12
0.700	4.9E-02	1.4E+10	4.1E+04	8.2E-05	00:14
0.800	3.8E-02	1.0E+10	3.2E+04	6.3E-05	00:16
0.900	3.0E-02	8.3E+09	2.5E+04	5.0E-05	00:18
1.000	2.5E-02	6.8E+09	2.0E+04	4.0E-05	00:20
2.000	6.5E-03	1.8E+09	5.4E+03	1.1E-05	00:40
4.000	1.9E-03	5.3E+08	1.6E+03	3.1E-06	01:20
6.000	9.6E-04	2.7E+08	8.0E+02	1.6E-06	02:01
8.000	6.1E-04	1.7E+08	5.1E+02	1.0E-06	02:41
10.000	4.4E-04	1.2E+08	3.7E+02	7.2E-07	03:21

6.2.3. Four-module case

HotSpot Version 3.1.2 General Plume

Source Material : Cs-137 F 30.0y
Material-at-Risk (MAR) : 4.0000E+13 Bq
Damage Ratio (DR) : 1.00
Airborne Fraction (ARF) : 1.000
Respirable Fraction (RF) : 1.000

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Leakpath Factor (LPF) : 1.000
Respirable Source Term : 4.00E+13 Bq
Non-respirable Source Term : 0.00E+00 Bq
Effective Release Height : 0.00 m
Wind Speed (h=10 m) : 2.00 m/s
Wind Speed (h=H-eff) : 0.83 m/s
Stability Class : F
Respirable Dep. Vel. : 0.30 cm/s
Non-respirable Dep. Vel. : 8.00 cm/s
Receptor Height : 1.5 m
Inversion Layer Height : None
Sample Time : 10.000 min
Breathing Rate : 3.33E-04 m3/sec
Distance Coordinates : All distances are on the Plume Centerline

Maximum Dose Distance : 0.010 km
Maximum TED : 4.85E+02 Sv
Inner Contour Dose : 0.250 Sv
Middle Contour Dose : **0.050 Sv**
Outer Contour Dose : 0.010 Sv
Exceeds Inner Dose Out To : 0.44 km
Exceeds Middle Dose Out To : **0.98 km**
Exceeds Outer Dose Out To : 2.3 km

FGR-13 Dose Conversion Data - Total Effective Dose (TED)

Include Plume Passage Inhalation and Submersion
Include Ground Shine (Weathering Correction Factor : None)
Include Resuspension (Resuspension Factor : Maxwell-Anspaugh)
Exposure Window:(Start: 0.00 days; Duration: 14.00 days) [100% stay time].
Initial Deposition and Dose Rate shown
Ground Roughness Correction Factor: 1.000

DISTANCE km	T E D (Sv)	RESPIRABLE TIME-INTEGRATED AIR CONCENTRATION (Bq-sec)/m3	GROUND SURFACE DEPOSITION (kBq/m2)	GROUND SHINE DOSE RATE (Sv/hr)	ARRIVAL TIME (hour:min)
0.030	4.5E+01	1.5E+11	6.6E+07	1.3E-01	<00:01
0.100	4.9E+00	1.0E+12	4.9E+06	9.7E-03	00:02
0.200	1.3E+00	3.3E+11	1.1E+06	2.2E-03	00:04
0.300	5.6E-01	1.5E+11	4.8E+05	9.4E-04	00:06
0.400	3.1E-01	8.4E+10	2.6E+05	5.1E-04	00:08
0.500	2.0E-01	5.3E+10	1.6E+05	3.2E-04	00:10
0.600	1.3E-01	3.7E+10	1.1E+05	2.2E-04	00:12
0.700	9.9E-02	2.7E+10	8.3E+04	1.6E-04	00:14
0.800	7.6E-02	2.1E+10	6.3E+04	1.3E-04	00:16

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0.900	6.0E-02	1.7E+10	5.0E+04	9.9E-05	00:18
1.000	4.9E-02	1.4E+10	4.1E+04	8.1E-05	00:20
2.000	1.3E-02	3.6E+09	1.1E+04	2.1E-05	00:40
4.000	3.8E-03	1.1E+09	3.2E+03	6.2E-06	01:20
6.000	1.9E-03	5.3E+08	1.6E+03	3.2E-06	02:01
8.000	1.2E-03	3.4E+08	1.0E+03	2.0E-06	02:41
10.000	8.8E-04	2.4E+08	7.3E+02	1.4E-06	03:21

6.2.4. Six-module case

HotSpot Version 3.1.2 General Plume

Source Material : Cs-137 F 30.0y
Material-at-Risk (MAR) : 6.0000E+13 Bq
Damage Ratio (DR) : 1.00
Airborne Fraction (ARF) : 1.000
Respirable Fraction (RF) : 1.000
Leakpath Factor (LPF) : 1.000
Respirable Source Term : 6.00E+13 Bq
Non-respirable Source Term : 0.00E+00 Bq
Effective Release Height : 0.00 m
Wind Speed (h=10 m) : 2.00 m/s
Wind Speed (h=H-eff) : 0.83 m/s
Stability Class : F
Respirable Dep. Vel. : 0.30 cm/s
Non-respirable Dep. Vel. : 8.00 cm/s
Receptor Height : 1.5 m
Inversion Layer Height : None
Sample Time : 10.000 min
Breathing Rate : 3.33E-04 m3/sec
Distance Coordinates : All distances are on the Plume Centerline

Maximum Dose Distance : 0.010 km
Maximum TED : 7.27E+02 Sv
Inner Contour Dose : 0.250 Sv
Middle Contour Dose : **0.050 Sv**
Outer Contour Dose : 0.010 Sv
Exceeds Inner Dose Out To : 0.54 km
Exceeds Middle Dose Out To : **1.2 km**
Exceeds Outer Dose Out To : 2.9 km

FGR-13 Dose Conversion Data - Total Effective Dose (TED)

Include Plume Passage Inhalation and Submersion
Include Ground Shine (Weathering Correction Factor : None)
Include Resuspension (Resuspension Factor : Maxwell-Anspaugh)
Exposure Window:(Start: 0.00 days; Duration: 14.00 days) [100% stay time].
Initial Deposition and Dose Rate shown

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Ground Roughness Correction Factor: 1.000

DISTANCE	RESPIRABLE TED	TIME-INTEGRATED AIR CONCENTRATION	GROUND SURFACE DEPOSITION	GROUND SHINE DOSE RATE	ARRIVAL TIME
km	(Sv)	(Bq-sec)/m3	(kBq/m2)	(Sv/hr)	(hour:min)
0.030	6.7E+01	2.3E+11	9.9E+07	2.0E-01	<00:01
0.100	7.4E+00	1.5E+12	7.3E+06	1.4E-02	00:02
0.200	1.9E+00	4.9E+11	1.7E+06	3.3E-03	00:04
0.300	8.4E-01	2.2E+11	7.1E+05	1.4E-03	00:06
0.400	4.6E-01	1.3E+11	3.9E+05	7.7E-04	00:08
0.500	2.9E-01	8.0E+10	2.5E+05	4.9E-04	00:10
0.600	2.0E-01	5.6E+10	1.7E+05	3.3E-04	00:12
0.700	1.5E-01	4.1E+10	1.2E+05	2.5E-04	00:14
0.800	1.1E-01	3.1E+10	9.5E+04	1.9E-04	00:16
0.900	9.0E-02	2.5E+10	7.5E+04	1.5E-04	00:18
1.000	7.4E-02	2.0E+10	6.1E+04	1.2E-04	00:20
2.000	1.9E-02	5.4E+09	1.6E+04	3.2E-05	00:40
4.000	5.7E-03	1.6E+09	4.7E+03	9.4E-06	01:20
6.000	2.9E-03	8.0E+08	2.4E+03	4.7E-06	02:01
8.000	1.8E-03	5.1E+08	1.5E+03	3.0E-06	02:41
10.000	1.3E-03	3.7E+08	1.1E+03	2.2E-06	03:21

6.2.5. Eight-module case

HotSpot Version 3.1.2 General Plume

Source Material : Cs-137 F 30.0y
Material-at-Risk (MAR) : 8.0000E+13 Bq
Damage Ratio (DR) : 1.00
Airborne Fraction (ARF) : 1.000
Respirable Fraction (RF) : 1.000
Leakpath Factor (LPF) : 1.000
Respirable Source Term : 8.00E+13 Bq
Non-respirable Source Term : 0.00E+00 Bq
Effective Release Height : 0.00 m
Wind Speed (h=10 m) : 2.00 m/s
Wind Speed (h=H-eff) : 0.83 m/s
Stability Class : F
Respirable Dep. Vel. : 0.30 cm/s
Non-respirable Dep. Vel. : 8.00 cm/s
Receptor Height : 1.5 m
Inversion Layer Height : None
Sample Time : 10.000 min
Breathing Rate : 3.33E-04 m3/sec
Distance Coordinates : All distances are on the Plume Centerline

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Maximum Dose Distance : 0.010 km
Maximum TED : 9.70E+02 Sv
Inner Contour Dose : 0.250 Sv
Middle Contour Dose : **0.050 Sv**
Outer Contour Dose : 0.010 Sv
Exceeds Inner Dose Out To : 0.62 km
Exceeds Middle Dose Out To : **1.4 km**
Exceeds Outer Dose Out To : 3.4 km

FGR-13 Dose Conversion Data - Total Effective Dose (TED)

Include Plume Passage Inhalation and Submersion
Include Ground Shine (Weathering Correction Factor : None)
Include Resuspension (Resuspension Factor : Maxwell-Anspaugh)
Exposure Window:(Start: 0.00 days; Duration: 14.00 days) [100% stay time].
Initial Deposition and Dose Rate shown
Ground Roughness Correction Factor: 1.000

DISTANCE km	T E D (Sv)	RESPIRABLE TIME-INTEGRATED AIR CONCENTRATION (Bq-sec)/m3	GROUND SURFACE DEPOSITION (kBq/m2)	GROUND SHINE DOSE RATE (Sv/hr)	ARRIVAL TIME (hour:min)
0.030	9.0E+01	3.0E+11	1.3E+08	2.6E-01	<00:01
0.100	9.8E+00	2.0E+12	9.8E+06	1.9E-02	00:02
0.200	2.5E+00	6.6E+11	2.2E+06	4.4E-03	00:04
0.300	1.1E+00	3.0E+11	9.5E+05	1.9E-03	00:06
0.400	6.2E-01	1.7E+11	5.2E+05	1.0E-03	00:08
0.500	3.9E-01	1.1E+11	3.3E+05	6.5E-04	00:10
0.600	2.7E-01	7.4E+10	2.3E+05	4.5E-04	00:12
0.700	2.0E-01	5.4E+10	1.7E+05	3.3E-04	00:14
0.800	1.5E-01	4.2E+10	1.3E+05	2.5E-04	00:16
0.900	1.2E-01	3.3E+10	1.0E+05	2.0E-04	00:18
1.000	9.8E-02	2.7E+10	8.2E+04	1.6E-04	00:20
2.000	2.6E-02	7.2E+09	2.2E+04	4.3E-05	00:40
4.000	7.6E-03	2.1E+09	6.3E+03	1.2E-05	01:20
6.000	3.8E-03	1.1E+09	3.2E+03	6.3E-06	02:01
8.000	2.4E-03	6.8E+08	2.0E+03	4.0E-06	02:41
10.000	1.8E-03	4.9E+08	1.5E+03	2.9E-06	03:21

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