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Review of SMR siting and emergency preparedness

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RESEARCH REPORT

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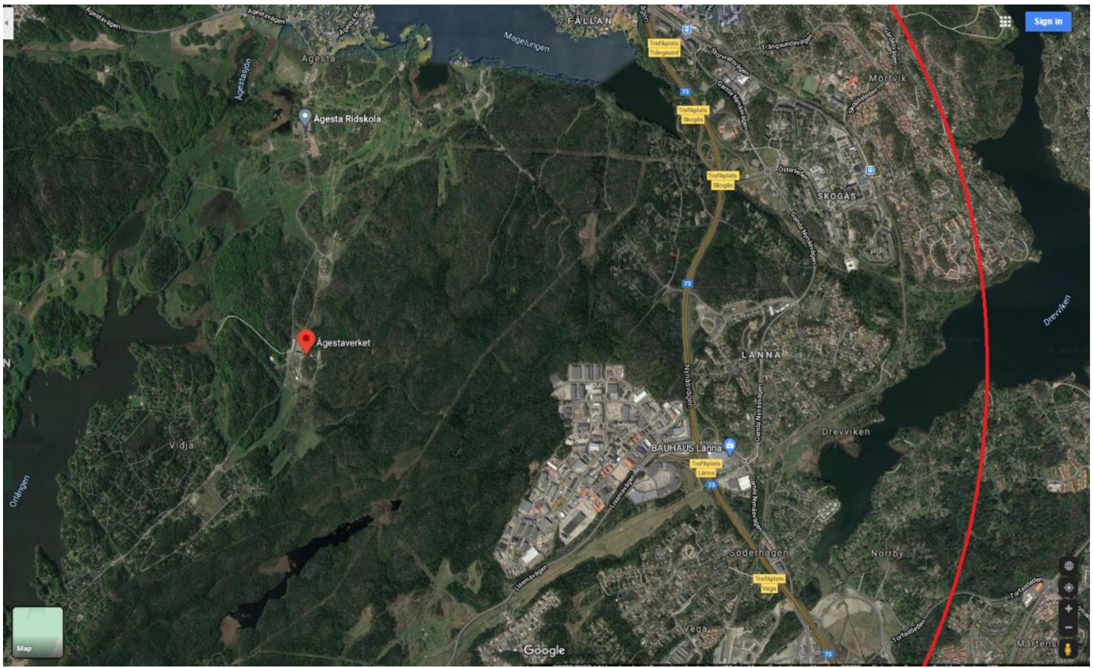

(Ågesta site, from Google Maps)

Review of SMR siting and emergency preparedness

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beyond the obvious

Report's title

Review of SMR siting and emergency preparedness

Customer, contact person, address EcoSMR Consortium	**Order reference** EcoSMR
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Summary

Small modular reactors (SMRs) have been a 'hot topic' of discussion in recent years, as they offer some potential advantages, like improved safety features and reduced capital costs per unit. A central questions in SMR deployment is the emergency planning zone (EPZ), whose size determines, to a large extent, where such plants can be located. This is important particularly for heat applications, as heat should not be transferred over long distances.

This work considers licensing issues of SMR plants through current international (IAEA, national regulators) developments in appropriate EPR (emergency preparedness and response) for SMRs. It is also proposed to study the problem numerically by a possibly definitive, but laborious & 'brute-force' approach by full-scope PSA to produce the basis (level 3: offsite consequences, like dose vs. distance) for choosing justified sizes of EPZ zones.

This review report contains a general literature survey on the SMR EPZ problem, but is also based on the author's work (2016-2018) in the GENXFIN project (in SAFIR2018) and in the IAEA CRP (coordinated research project) I31029 'Determining the technical basis for EPZ for SMR deployment' (2018-2021). Some information / insight also comes from participation in various IAEA Technical Meetings (TM), particularly the Feb 2017 TM on NGR EPZs, and the Sep 2020 TM on NGR and EPR (NGR = next generation reactor). Other relevant meetings were the Apr 2019 TM on Advances in EPR arrangements, and the Dec 2019 Regional workshop on EPR for SMRs.

In Finland, a MEAE-set working group gave their final report 27 Aug 2020 on the coming reform of nuclear regulation. Earlier in 2020 STUK gave out an overview report on Preconditions for the safe use of SMRs. For assessing the possibility of SMRs in Finland, the licensing process, licensing requirements and related safety issues should be studied in more detail. In the author's (of this work) view this includes particularly passive safety systems and further evaluation of possible offsite consequences, down to doses and health effects.

Confidentiality	Public

Espoo 1.2.2022

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Preface

The main objective of the EcoSMR project (Industrial / business ecosystem for small modular reactors in Finland) is to enable Finnish companies to participate in SMR (small modular reactor) markets by consulting, supply chains and SMR integration for new applications. EcoSMR also prepares for a domestic design of a district heating reactor. VTT launched a project to develop the Low Temperature District Heating and Desalination Reactor (LDR) in 2020. Generally, EcoSMR can increase scientific and technological knowledge of new nuclear energy technologies in Finland. EcoSMR has web pages at https://www.ecosmr.fi/ and contact person Ville Tulkki (ville.tulkki@vtt.fi).

VTT has gathered together a group of Finnish utilities & other companies interested in SMR development: VTT, LUT, AFRY, Clenercon, EnviroCase, Fortum (Loviisa owner), HELEN (Helsinki Energy), Refinec, RockPlan, TVO (Olkiluoto owner) and Vantaan Energia. Business Finland (a funding organization for Finnish export industry) granted money for research at VTT and LUT, and for development projects at several companies. The group wants to create a domestic branch of industry capable of manufacturing most components.

EcoSMR focus is on licensing, heat use of nuclear energy, and business models:

- Not possible that each municipal heat provider be a nuclear utility
- ’Heat as a service’ model possible
- Specialized competence & resources will be needed.
- International designs are generally too large and would only be useful for the capital region in Finland.

Research in EcoSMR includes licensing, regulations, design criteria, heat use of small reactors, business models, case studies and ecosystem activities. In this case, ecosystem involves sharing of expertise between companies and establishing also international connections, like participation in international working groups and meetings. This report considers licensing issues of small modular reactors: the current international (IAEA, national regulators) developments in determining the sizes of emergency preparedness and response (EPR) zones of SMRs.

Espoo, 11.1.2021

Mikko Ilvonen

(Cover page image from Google Maps: Ågesta site near Stockholm with 5 km distance shown)

Contents

1. Introduction

Quite a lot of international interest has been in SMRs (small modular reactors) in the recent years. Even in Finland, the electric utility Fortum and several cities have expressed their interest. An SMR could be a small initial investment and be used for e.g. combined generation of electricity and heat. However, the questions of licensing, siting and emergency planning should receive more rigorous consideration, among many other technical and financial issues.

Heating reactors in particular should be located near the customers to avoid heat transfer losses, but at present this is not possible because of emergency preparedness considerations. The PAZ (Precautionary Action Zone, or ‘suojavyöhyke’ / ‘protection zone’ in Finland) is 5 km from the plant. The PAZ shall not be densely populated; protective actions shall be implemented immediately when a general emergency is declared. The UPZ (Urgent Protective actions planning Zone, or ‘varautumisalue’ / ‘preparedness zone’) is 20 km from the plant. In the UPZ, there must be preparedness to perform protective actions, i.e. requirement is to demonstrate that protective actions can be performed efficiently if needed. PAZ and UPZ together form the Emergency Planning Zone (EPZ).

Presently in Finland, it is not possible to site a heating reactor in a city, but in near future the detailed requirements may change in the coming regulatory reform in Finland. Launching of legislative preparations was officially announced by TEM / MEAE (Ministry of Economic Affairs and Employment in Finland) in December 2021 to reform the Nuclear Energy Act. Obviously however, the overall required level of safety should remain at least the same as before. Generally the safety of a small heat-only reactor should be easier to prove than that of a large NPP.

For siting even a small heating reactor in city area, Large Early Release should be practically impossible (LERF frequency very low). Then a smaller emergency planning zone (EPZ) could be possible, together with possible other differences from present large NPP EPZ:

- Graded approach when selecting / planning relevant types of protective actions?
- Combining actions with other, ‘conventional’ emergency planning?
- Centralized emergency resources for several district heating (DH) stations?

The main point of this report is not the licensing of SMRs under current regulation, but rather a (very superficial) survey into the question, if there could be SMR-specific regulations, and if so, how the emergency management requirements could be defined. Specifically, could the EPZ requirement (in Finland currently 5 / 20 km) be substantially smaller? Where in the environment could STUK dose limits be exceeded, and what would be the expected frequencies of exceeding limits?

A good methodology for EPZ determination (for SMR, or more general) should be traceable (to see how the results were produced), flexible (different solutions, differences of countries), technology-neutral (same method for LWR, HTGR, etc.) and represent the current state of the art (e.g. latest international requirements).

Proposed strategy to determine right-sized EPZ

It is suggested in this report that the definitive, rigorous way to solve such questions would be the complete PSA/PRA (Probabilistic Safety / Risk Analysis), including levels 1, 2 and 3 (offsite consequences), but also augmented with deterministic analyses to produce more detailed information according to the actual phenomena. Such procedure has been mentioned in several references, like Mancini et al. (2014), IAEA SMR Regulators’ Forum EPZ WG (working group on EPZ), or the licensing process in Canada, where the size of the EPZ around any new NPP is basically flexible. However, it must be admitted that the full-scope PSA is a huge work, both to perform and also to review by the regulator. Deterministic calculations of a few ‘bounding sequences’ are much easier.

In PSA1-2-3, there are basically a huge number of different atmospheric radioactive releases (from level 1 & 2 event branchings) with their frequencies of occurrence, and each release can be dispersed by a huge number of different weather conditions (affecting different releases differently). Offsite doses received without protective countermeasures should be compared to dose limits, and then, if necessary, checked with the countermeasures, if the doses can be reduced sufficiently. If this is not possible, for example due to too much population in the area, then the area should be low population zone.

The PSA study would usually be specific to a certain plant design & siting combination. To have more general rules, there would have to be several designs and sites with assumed site-specific external events etc. Such ‘enveloping’ idea was used in the ESPA (early site permit application) of TVA for the Clinch River Site in the US in 2016.

Even a huge number of calculations on level 3 is relatively straightforward to perform with a fast atmospheric dispersion and dose assessment model, like the ARANO code developed at VTT. It has built-in code for several source terms and weather conditions with their frequencies, or could also be used as a subroutine call from an outside control loop.

In a comprehensive EPZ sizing study, one should ultimately answer questions like ‘How far from the nuclear plant must I be so that my frequency of receiving more than 10 mSv of effective dose in 2 days will be lower than x per reactor-year?’ Where the value x depends on the question ‘How safe is safe enough?’

The practical steps in determining a ‘right-sized’ EPZ can be listed like the following, for a single atmospheric source term assumption:

1. Atmospheric release source term, with nuclide-wise Bq & temporal behaviour, also at least effective release height
2. All the criteria for needing to perform certain protective measures, like in Finland ‘Evacuate if more than 10 mSv / 2 days’
3. Ideally, real weather data for a site (or several sites) for a period of several years
4. Offsite dose assessment with computer code / tool X to calculate doses according to the criteria (exposure pathway & integration time)
5. Look at dose vs. distance, ideally e.g. 95 % percentile from all weather cases, to see up to which distance the criterion was exceeded
6. We might have different zone sizes for different protective measures - not plan for everything up to the same distance

Finally, we may distinguish ‘two levels’ in deciding about EPZ: the ‘technical’ level (1-6 above) and also a ‘meta level’, meaning how to use the results (how much ‘marginal’ to include for technically not so straightforward considerations):

- Have as the primary basis the rigorously calculated offsite doses from steps 1-6 above, compared with dose criteria;
- Make the final decisions possibly taking into account population & traffic considerations, public fears, low-probability events, etc. etc. secondary considerations (by the regulator).

Especially interesting published work has been done by Mancini et al. (2014), considered in more detail further below, who claim to combine ‘probabilistic, deterministic, and risk management methods that would support licensing with reduced emergency planning requirements’. In addition to PRA, they have used the codes MAAP, RADTRAD and RODOS.

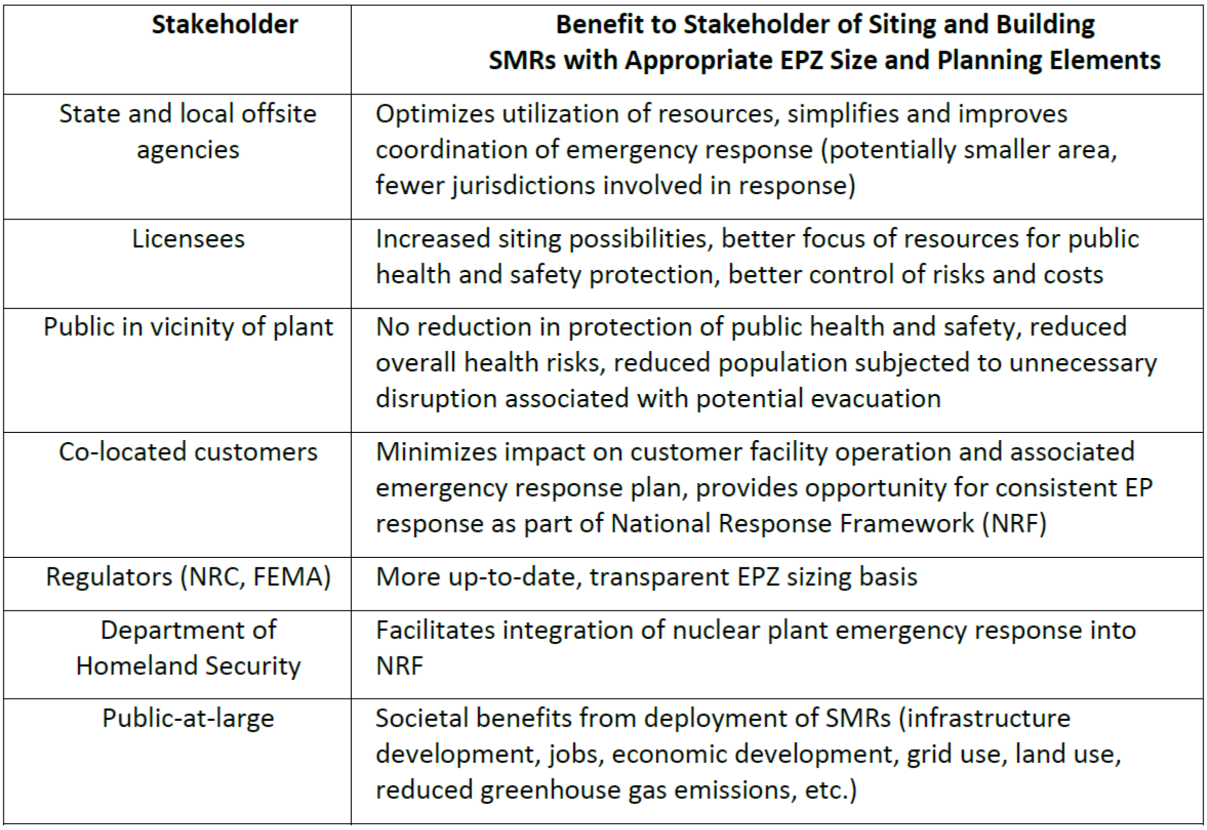
Potential benefits of a reduced EPZ size

There are many good reasons why particularly the industry (SMR plant providers, operators and heat / electricity users) would like to see reduced EPZ sizes:

- More potential sites, like former coal-burning plants, near cities
- Less heat losses when supplying heat to industry or district heating
- Simpler & cheaper emergency planning
- More straightforward licensing for both applicant and regulator

The industry-side NEI (2013) 'White Paper' on SMR EPZ lists their reasons to consider more appropriate EPZ sizing for SMRs - see Table 1 below.

Table 1. Potential benefits to SMR stakeholders of siting SMRs with appropriate EPZ size. List expresses the industry point of view. Source of Table: NEI (2013).



Stakeholder	**Benefit to Stakeholder of Siting and Building SMRs with Appropriate EPZ Size and Planning Elements**
State and local offsite agencies	Optimizes utilization of resources, simplifies and improves coordination of emergency response (potentially smaller area, fewer jurisdictions involved in response)
Licensees	Increased siting possibilities, better focus of resources for public health and safety protection, better control of risks and costs
Public in vicinity of plant	No reduction in protection of public health and safety, reduced overall health risks, reduced population subjected to unnecessary disruption associated with potential evacuation
Co-located customers	Minimizes impact on customer facility operation and associated emergency response plan, provides opportunity for consistent EP response as part of National Response Framework (NRF)
Regulators (NRC, FEMA)	More up-to-date, transparent EPZ sizing basis
Department of Homeland Security	Facilitates integration of nuclear plant emergency response into NRF
Public-at-large	Societal benefits from deployment of SMRs (infrastructure development, jobs, economic development, grid use, land use, reduced greenhouse gas emissions, etc.)

Just one year after the NEI SMR EPZ methodology proposal, the Idaho National Laboratory (INL, 2014) discussed the benefits and costs of SMR EPZ sizing in a discussion paper of the ART (Advanced Reactor Technologies) program, prepared for the DoE. They point out the cost of EPZ size - see Figure 1 below.

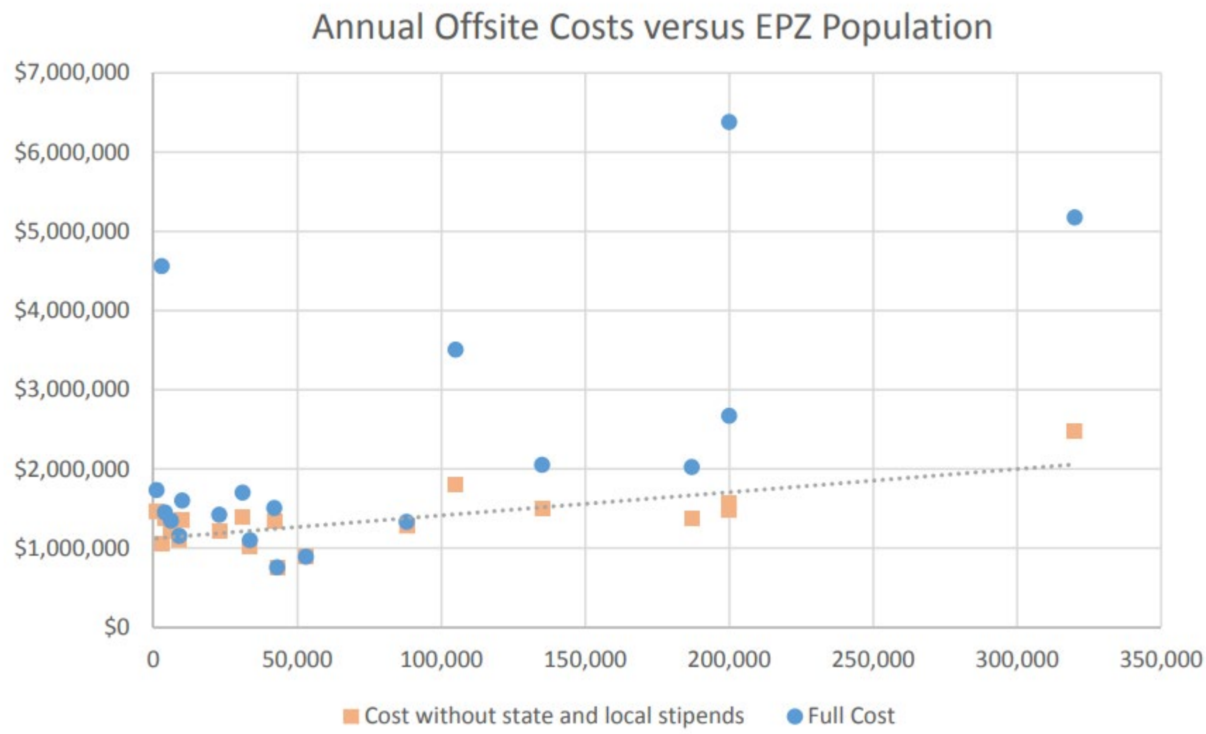


Figure 1. Annual cost of offsite emergency preparedness vs. EPZ population in the US. Source of Figure: INL (2014).

The discussion brought to public by e.g. the above examples, NEI (2013) and INL (2014) publications show the strong 'pressure' to reduce the EPZ size for SMR plants. In this report, the central theme is

- Is it safe to reduce the EPZ sizes, how safe is safe enough, and what would be the scientifically justified way of calculating exactly adequate EPZ sizes?

NEI (2013) crystallizes emergency planning by stating that 'the primary objective of EP as indicated in NUREG-0396 is to produce dose savings for a spectrum of accidents that could potentially lead to offsite doses in excess of the EPA protective action guides (PAGs)'.

Potential justification of reduced EPZ

The smaller core / thermal power of SMRs practically means that the inventory of the fission products is smaller. Like residual decay heat, this inventory is roughly proportional to the reactor power (but also to burnup). So also the environmental consequences of an atmospheric radioactive release from an SMR are expected to be less severe than from typical present LWRs. With a smaller core, also cooling of residual decay heat is easier. These facts could be utilized in terms of smaller emergency preparedness zones (EPZ) and reduced radiation shielding. This would mean lower costs to the licensee in maintaining offsite emergency preparedness and in fulfilling nuclear liability requirements (cost of insurance). However, one site could contain several SMR units, with the total source term as large as in a large power reactor, if there is a common failure.

To determine the proper size of the EPZ for an SMR or any other NPP, with particular design and safety features, the postulated atmospheric source terms have to be determined. (Radioactive releases into aqueous pathways may cause longer-term effects.) Then applying international safety criteria, the distances for the EPZs can to be obtained. Atmospheric dispersion conditions affect significantly the offsite doses. If weather data is based on annual data, a probabilistic approach to doses can be adopted.

Exposure modes considered should include the relevant dose pathways: external radiation from the plume, internal from inhalation, and external radiation from the deposition on the ground. The outcome is then recommendations, with proper justification, on the emergency planning and response for the plant. This includes EPZs (emergency planning zones) based on the international safety standards.

EU ans IAEA cooperation

This report also contains some brief information on SMRs in general, international (IAEA) safety standards, regulatory processes involving SMRs in some countries, international collaboration where VTT participates, and a general-level description of a definitive, rigorous PSA-based process of determining justified EPZ sizes:

- Literature study with discussion on the required size of SMR emergency preparedness and response (EPR) distances or zones
    - Plant providers usually try to justify small zones;
    - Regulators, in many cases, have not made it clear if and with exactly what justification the zones could be approved during the licensing process.
- Reduced EPZ size could be realized taking advantage of the smaller reactor core radioactive inventories and the more advanced safety features of the new plants.
- Smaller EPR distances around new types of nuclear power plants and the possibility to completely do away with EPR arrangements remain controversial.
- Ideally, a rigorous analysis of the EPR distances of SMRs should be made based on actual radioactive inventories, modelled DF (leak path decontamination factors) and resulting atmospheric release source terms, as well as a computational assessment of doses and their comparison with international action levels for radiological countermeasures.
- The main question is how far from an SMR plant the PAZ (precautionary action zone) and UPZ (urgent protective action planning zone) should reach.
- Particular importance if the SMR plants are to be used as a local source of heat for cities and industry
- This literature study briefly covers also
    - Emergency planning provided by SMR plant vendors
    - Examples of regulatory policies internationally
    - Analyses on the topic made by research institutes and consulting firms.

VTT Nuclear Energy (i.e. the author) has participated in several IAEA CRPs (Coordinated Research Projects) and TMs (Technical Meetings) related to SMR EPZ determination, for example I31029 ('Technical basis for EPZ for SMR) and J15002 ('Effective use of dose projection tools'). The results from those will be described in more detail further down in this report. Particularly the I31029 ended in 2021, after a half-year prolongation and a total of 4 RCM Research Coordination Meetings, but bore fruit in the sense that the agency is now asking participants to contribute to the writing of an IAEA TECDOC document on SMR EPZ determination guidelines. That is expected to bring the CRP work to a major common conclusion in 2022.

In December 2021 there was an IAEA webinar on 'Applicability of IAEA Safety Standards to the Design of Novel Advanced Reactors including SMRs'. The IAEA has performed a high-level mapping of applicability of the IAEA safety standards to Novel Advanced Reactors (NARs) including Small Modular Reactors (SMRs), high temperature gas cooled reactors (HTGR), sodium fast reactors (SFR), lead fast reactors (LFR), molten salt reactors (MSR), marine based SMRs and micro reactors. In this webinar, focus was on design safety (DiD levels 1-4) and requirements could be either applicable or not. On the other hand, in previous IAEA meetings it was clearly concluded that GSR Part 7 (General Safety Requirements on EP&R, Emergency Preparedness & Response, DiD level 5) is well applicable to all kinds of reactors, both old and new.

In 2020, VTT Nuclear Energy was participating in several SMR-related EU projects: ELSMOR, McSAFER and EEC SMART. In ELSMOR, the consortium is using methods for robust safety assessments and studying selected safety features of light water LW-SMRs, e.g. core cooling and containment functions to prevent early release. Regarding the present EPZ problem, ELSMOR produced a very interesting deliverable in 2021 about ‘Determination of Emergency Planning Zones and Scaling Acceptance Criteria for Downsized Nuclear Power Plants’ by JRC researcher de la Rosa Blul. McSAFER is studying experimental thermal-hydraulics (TH) validation and suitability of tools of different fidelity (assembly / pin / sub-pin level) for modelling SMR transient scenarios with multiple concepts being modelled (Nuward / France, SMART / Korea, NuScale / USA, CAREM / Argentina). ECC SMART has a European-Canadian-Chinese consortium studying identification and feasibility of safety features of intrinsically & passively safe SCW-SMR (Super-Critical Water SMR). Focus is on behavior of materials in the SCW environment and under irradiation, validation of the codes and design of the reactor core. In October 2021, VTT participated in submitting an EU project proposal SASPAM_SA, where VTT coordinates WP6 (Work Package 6, Characterization of iPWR EPZ). Two kinds of iPWR (integral Pressurized Water Reactor) are proposed to be studied: one with a submerged containment and electric power of about 60 MWe, and another with use of several passive systems, a dry containment and an electric power of about 300 MWe.

Some historical view

The use of small / SMR-sized reactors, even for heat generation, is actually not a new idea in Finland or Sweden. Some historical projects were described by (Leppänen 2018). The Ågesta plant, in Farsta suburb of Stockholm, operated in 1964-74 using natural uranium & heavy water moderator. The mode was co-generation producing 10-12 MW electricity and 55-68 MW district heat. The cover picture of this report shows satellite view of Farsta with a hypothetical zone of 5 km radius around the Ågesta plant. At the same time in Finland, there was a national project on viability of nuclear district heating in 1971-73. VTT (coordinator), Ekono, IVO (now Fortum), TVO, and IFA (Norway, now IFE) produced a conceptual design of a reference 100 MW LWR reactor plant. The costs were evaluated, assuming 12 largest cities in Finland. After that, SECURE (‘Safe Environmentally Clean Urban Reactor’) was designed by Swedish-Finnish collaboration in 1976-77, including ASEA-Atom, Atomenergi, Finnatom & VTT. SECURE had 200 MWth output at 95 deg C and was to be sited also close to densely populated areas. Because of its inherent safety features with nitrogen and boron, ‘no large EPZ would be needed’.

To be on a sound basis in setting EPZ distances, ideally a lot of knowledge about potential reactor accident consequences is needed: expected frequencies of various accidents at various kinds of nuclear power plants at various locations, and their consequences (doses, health effects) at various distances from the plant. An early example (NRC 1975) of such ‘knowledge base’ is the WASH-1400 Reactor Safety Study (‘the Rasmussen report’). It used an event tree approach to identify possible accident sequences, starting the eventually wide-spread use of PRA (Probabilistic Risk Assessment) in nuclear safety. Current EPZ regulation in the US (10 miles cloud, 50 miles ingestion) derives from dose calculations based on WASH-1400 sequences and source terms.

A few years later in 1978, an essential document regarding EPZ appeared: the NUREG-0396, giving basis for radiological emergency response planning. Therein reference was made to results of WASH-1400. To cite the 0396 (p. I-37): ‘Given a core melt accident, there is about a 70 % chance of exceeding the PAG doses at 2 miles, a 40 % chance at 5 miles, and a 30 % chance at 10 miles from a power plant. That is, the probability of exceeding PAG doses at 10 miles is 1.5e-5 per reactor-year (one chance in 50000 per reactor-year) from the Reactor Safety Study analysis. Based in part upon the above information the Task Force judged that a 10 mile plume EPZ would be appropriate to deal with core melt accidents.’

Detailed review of NUREG-0396 reveals a lot of EPZ-related wisdom from more than four decades ago. Even when the 0396 document is for 1970s US LWRs, many of the same ideas / principles could be applied to the present SMR EPZ problem:

- Look at a spectrum of accidents (instead of e.g. just one so-called ‘worst case’), consider many plant types & locations (then 129 NPPs).
- Protection measures will be affected by distances, timing and nuclide content of the plume.
- What are the measures to plan for emergency preparedness?

- Some critical human organs (red marrow, thyroid, inhalation exposure pathway) will most probably define action distances.
- Low wind speeds will cause high doses, but on the other hand, provide more time to act before the plume front arrives.
- Results from simple Gaussian dispersion models may be unreliable because of low wind direction persistence probabilities, later rainfall along the plume axis, etc.
- Time frames of accident & release progression will define if actions are needed as a precaution, or urgently after information is available.
- The probability rationale used in WASH-1400 has both pros and cons.
- There is always some remaining probability of needing radiological countermeasures even beyond the defined EPZ; but how big is that probability?
- Other than nuclear emergencies, like tornados, must be considered in two ways: As causing a nuclear emergency, and as making protection measures more difficult to implement.
- In any EPZ analysis, it is essential to list possible conservative (or not) assumptions.

2. Small Modular Reactor (SMR) typical features

A short introduction to common basic features of Small Modular Reactors (SMRs) is included here. For more complete description, see e.g. VTT-R-05548-16 [Hillberg et al. 2016]. The international development of emergency preparedness and response (EPR), as described in this report, is basically for any kind of SMRs, not only the LWR type.

The features of an SMR can be listed, partly by the WNA *[World Nuclear Association, 2016]*:

- Small power (< 300 MWe) and compact architecture
- Smaller core radioactive inventory, because of small power
- More heat transfer surface per unit of power
- Possibly underground or underwater location of the reactor unit
- Possibly multiple units at the site (more than for large NPPs)
- Passive features and safety systems
- In-factory fabrication of a SMR unit
- Replace economy of size with economy of modular/serial production in same factory
- Decrease the initial investment needed
- Lower requirement for access to cooling water
- Suitability also for remote regions & isolated / low capacity grids
- Easier daily load following (even with intermittent energy sources)
- Possibly whole reactor module removal for decommissioning at the end of the lifetime
- Applications: Electricity generation, district heating, cogeneration, water desalination, high temperature process heat for process industry, hydrogen production
- Output range suitable to existing heat and water distribution networks

The currently developed SMR concepts can essentially be divided to Gen III/III+ and GenIV designs *[Subki and Reitsma, 2014]*. LWR (light water reactor) is the most common NPP design in the world: there are around 437 reactors in operation and of them 357 are LWRs. Of these LWRs 273 are PWRs, so most operating experience has been acquired with PWR technology. LWR SMRs have a relatively low technological risk but the advanced designs may be smaller, simpler, with longer operation before refuelling *[Lokhov and Sozoniuk, 2016; Kollar, 2015]* and better possibility of fuel recycling. In Finland, GenIII type of SMRs are more likely to be deployed in commercial use in near term.

Extensive basic data on SMRs in tabulated form can be found in the ‘SMR Book’ of (IAEA 2020). SMRs of the LWR type are generally the most mature kind of SMR. These include e.g. NuScale, Westinghouse SMR, mPower, SMART (Korea), CAREM-25 (Argentina), and KLT-40S (Russia) on the floating Akademik Lomonosov. From the licensing point of view, central questions include the consideration of passive safety systems, severe accidents (even when ‘ruled out’ by the plant provider) and the size of the emergency preparedness zone (EPZ) around the plant. Plant providers may be willing to suggest that no bigger EPZ than the site area is needed.

In some SMR designs, like NuScale, all safety-critical equipment, including the reactor and the fuel vessels, will be located underground, minimising the need for expensive physical defences. NuScale, with design of integral PWR or iPWR type, is aiming to build its first SMR plant in the US state Utah and also in Europe in the UK by 2029. For new reactor licensing in the US, there is a ‘Design Certification Review’ after which the design can be incorporated in later applications just by reference. Design certification does not include some site specific items and some potentially fast changing technology areas, but it should identify design acceptance criteria to be satisfied in later applications. In January 2017, the NRC received the DCA (Design Certification Application) from NuScale. According to news in January 2018, the NRC then concluded e.g. that the NuScale SMR design does not need backup electric power supply of Class 1E. The preliminary safety evaluation report (SER), after NRC staff review, was completed by the NRC in April 2018. Then the FSER (Final Safety Evaluation Report) was issued in August 2020, leading to SDA

(Standard Design Approval) in September 2020. In parallel with the DCA process, NuScale requested in December 2015 (with Revision 1 in March 2018) that the NRC review their design-specific plume exposure EPZ sizing methodology ('Methodology for establishing the technical basis for plume exposure emergency planning zones at NuScale small modular reactor plant sites').

Potential benefits of SMRs, compared with present day's typical large NPPs include the following *[Carelli, 2014; Subki and Reitsma, 2014; World Nuclear Association, 2016; Lokhov and Sozoniuk, 2014; Rowinski, 2015]*:

- Build many small similar units, license once (site / plant type), produce serially in a factory (enhancing quality), transport in one piece
- Shorter construction schedules, smaller initial investment, no very big components
- Lower grid capacity (like in developing countries) sufficient, smaller backup power need, possibly operation in own local isolated grid
- Load following: heat/electricity cogeneration, number of SMR units in production
- Smaller core radioactive inventory
- Possibly smaller size of the EPZ (Emergency Planning Zone)
- Easier decommissioning (modularity, small-sized units)
- Short unit-by-unit maintenance and refuelling, human resource management of teams; possible problems with currently maintained and power-producing units being located close to each other (more units than for large NPPs)

The practical main reasons why SMRs could be desirable to build are the reduction of the total capital costs of the projects and shortened construction schedules. Also the many enhanced safety features & simplified designs support their choice.

Hidayatullah et al. (2015) give a good, concise table of SMR features with their added values and expectations on the safety ('Table 3: Expectation on the safety of SMRs').

Ramana & Mian (2014) list four central, unresolved problems of nuclear power, that the SMR plant providers try to offer their solutions for:

- Economics: Large NPPs have become very slow and expensive to build. A big initial investment is required.
- Safety: Classic concern by the public and politicians, despite all the improvements made over the decades.
- Nuclear waste: Finland has a working solution by Posiva, but most countries do not.
- Proliferation: It is feared that NPPs may 'pave the way' to military use of U-235 or Pu-239.

3. Licensing issues of SMRs

The IAEA Regulators' forum defines licensing as 'The official process of authorization granted by the regulatory body to the applicant to have the responsibility for the siting, design, construction, commissioning, operation or decommissioning of a nuclear installation.'

Sainati et al. (2015) use IAEA terminology to classify the licensing approaches in various countries to prescriptive or performance-based:

- Prescriptive licensing: Most common, mostly based on DSA (deterministic safety assessment), with pre-defined norms and principles for materials, components etc. Efficient licensing for experienced operators. Licensing uncertainties / ambiguities may be reduced. In some countries, the codes and standards may be almost tailored to a specific reactor design.
- Performance-based licensing (goal setting): Sainati et al. (2015) mention the UK as an example. Risk-informed regulation & ALARA (as low as reasonably achievable) principle. More flexible for new reactor designs, but the regulator might make subjective decisions. Usually two parts: design certification & site certification.

Sainati et al. (2015) emphasize the duration and predictability of the licensing process (LP). They summarize that existing LPs could extend SMR construction times beyond the pure technical schedule undermining the overall economics.

In Finland, goal setting is mentioned in 'Rules for application' of e.g. YVL A.2 (Site for a nuclear facility): 'According to Section 7 r(3) of the Nuclear Energy Act, the safety requirements of the Radiation and Nuclear Safety Authority (STUK) are binding on the licensee, while preserving the licensee's right to propose an alternative procedure or solution to that provided for in the regulations. If the licensee can convincingly demonstrate that the proposed procedure or solution will implement safety standards in accordance with this Act, the Radiation and Nuclear Safety Authority (STUK) may approve a procedure or solution by which the safety level set forth is achieved.'

The NRC have included in their draft regulatory basis of 'Rulemaking for EP for SMRs and ONTs' (2017) also the possibility for performance-based approach to emergency preparedness. Currently prescriptive planning standards are the way in the US for large LWRs to meet EP requirements. According to Rahn et al. (1984), there were some standardization attempts (to achieve easier licensing) in the US as early as around 1980, but with little success. In the 2017 suggested approach, the licensee would have flexible choice of how to meet the stablished EP performance criteria.

Graded approach (GA) means that requirements should not be the same for all kinds of nuclear facilities, but rather depending on its size and other factors. The principle of GA has been applied to e.g. the FiR-1 research reactor in Finland. The IAEA Regulators' Forum GA WG (working group on graded approach) finds in their report that here are many questions remaining about appropriate ways to perform grading in design and safety analysis work, though it has been used in some form for a long time. The SMR industry is asking, what would be necessary to demonstrate that something is proven and looking for more objective-based regulatory approaches with less prescriptive requirements. The GA WG concludes that the IAEA does not prescribe any specific methodologies, but presents enough guidance to allow Member States to develop appropriate acceptance criteria under their own regulatory framework.

In Canada, GA means that the level of analysis, the depth of documentation and the scope of actions necessary to comply with requirements are commensurate with relative risk to health and the characteristics of the facility. For Finland, the GA WG reports that the principle of GA was added into the Finnish Nuclear Energy Act in the year 2013 (499/2013). Section 7a of the Act states that 'Safety requirements and measures to ensure the safety shall be sized and allocated proportionate to the use of nuclear energy risks.'

A brief introduction is given here to the possible licensing of SMRs by STUK in Finland and by NRC in the USA. The levels of the Defence-in-Depth concept are described and the 5th level (emergency preparedness and response, EPR) is emphasized. In the licensing process, the consequences of accidents can only be properly described after calculating also assessments of dispersion and radiation doses. The interested reader can refer to e.g. ‘Licensing’ in VTT-R-05548-16 (Hillberg et al. 2016) for more information.

The main strength of SMRs is modularity. The current Finnish licensing process was not made for modular licensing, or ‘reactor type approval’. The question could be compared with cars, which (on the contrary) may be sold in Finland after a certain model is approved. Of course, for a nuclear power plant one must also take into account the site-specific conditions. SMRs could benefit from developing an internationally applicable “Standard Design Certificate of Module” (SDCM) that would ensure the safety of the module design and lead to harmonization of nuclear licensing internationally *[Söderholm, 2013].* One central question is the usual manner, in which STUK also controls the design phase and supply chain in case-by-case licensing. In factory production of a number of modules, these actions may have already happened before a Finnish utility orders the plant from provider.

New reactor concepts are generally designed to eliminate as many vulnerabilities and initiating events of incidents and accidents as reasonable achievable. Safety improvements can be achieved by using inherent safety features and passive safety systems in new plant designs, and remembering also the lessons learnt from old NPPs. In most mature SMR designs, safety concepts are based on the DiD principle, so there should not be any fundamental reason why they could not be licensed in Finland. The common design principle of many interesting SMR designs, utilization of inherent safety and passive safety systems, can be found very clearly in the Finnish regulations on nuclear safety, e.g. in YVL guide B.1: Safety Design of a Nuclear Power Plant. Issues of dispute (designer vs. regulator) may include the safety grade of systems and the independence of the DiD levels.

The Finnish licensing process is currently designed for large LWRs. This makes the licencing process quite rigid. For example, the design phase and supply chain should be looked at quite intensely by the regulator. The process does not take into account the different design features of SMRs like modularity and multi-reactor installations. However, there is no reason why SMRs could not be licensed in Finland if the Finnish requirements are met. In April 2018, Jorma Aurela of TEM gave a presentation in Finnish Nuclear Society’s ATS YG about the possible licensing of several SMR units (at the same time, as one ‘ensemble’) in Finland using the current framework (Decision-in-Principle, Construction permit, Operating license). It seems that there are clearly difficulties, though theoretically the mentioned licensing is possible.

4. Introduction to emergency preparedness

4.1 Emergency planning zones and distances

The EPZ (emergency planning zone) around an NPP is an area where the licensee (NPP operator) has responsibilities for alerting and protecting the offsite population if this is made necessary by a radiological hazard caused by the plant. For the inhabitants, it means basically being informed about such possible actions, but may seem alarming to some of them. In Finland, the YVL Guide A.2 (Site for a nuclear facility) defines prescriptively the zones: 0.5 -1 km (site area), appr. 5 km (PAZ, 'suojavyöhyke') and about 20 km (EPZ, 'varautumisalue').

Opponents of SMR point out the EPZ question, e.g. the UCS (Union of Concerned Scientists, 2013): 'SMRs could be located at former coal plants, at industrial sites to provide process heat, at military bases, or indeed in any densely populated area, without the burden of developing evacuation plans and evacuation time estimates, deploying and maintaining sirens, and, most notably, without notifying and educating the public about the need to evacuate.'

In the US, NRC regulations for siting of NPPs include (100.11) determination of an Exclusion Area (EA) controlled by the licensee, and a Low Population zone (LPZ), both in terms of individual radiation doses from a postulated fission product release. Historical explanation can be found e.g. in Rahn et al. (1984), p. 801. For emergency planning, there are two zones:

- 10 miles plume exposure pathway EPZ (short term)
- 50 miles ingestion exposure pathway EPZ (longer term)

After the TMI accident of March 1979, the responsibility of offsite emergency planning (legislation 44CFR350) was shifted to FEMA (Federal Emergency Management Aggency). Note also: The dose limits are set in the PAGs (Protective Action Guides) of the EPA (Environmental Protection Agency).

Internationally, the zones related to EPR (emergency preparedness & response) and longer-term actions can be explained as follows. Detailed explanations are best found in the EPR-NPP_PPA of IAEA (2013):

- PAZ (precautionary action zone): Preparedness for precautionary urgent protective actions (before release or shortly after it begins) to reduce the risk of severe deterministic effects. Extends e.g. to 5 km from a typical large power reactor.
- UPZ (urgent protective action planning zone): Preparedness for urgent protective actions to be taken promptly to avert offsite doses. Extends to 25 km from a typical large NPP. Note: In Finland, the term 'Emergency planning zone' (EPZ), extending to 20 km from reactor, was usually used.
- LPZ (longer-term protective action zone), also called FRPZ (food restriction planning zone): Preparedness for protective actions to reduce the long-term dose (stochastic health effects from groundshine and ingestion of local food). The LPZ may extend to 300 km from a large NPP.
- EPD (extended planning distance): Monitor the situation to find areas in which response actions would be needed within the time period 1 d to a few weeks. E.g. 100 km.
- ICPD (ingestion and commodities planning distance): Reduce stochastic effects due to contaminated food, milk and drinking water, and commodities other than food. E.g. 300 km.

According to the IAEA Regulators' Forum EPZ WG, the longer distances EPD & ICPD are detrmined by surveys only after the release.

The Emergency Planning Zone (EPZ) is defined by the IAEA as follows:

An EPZ consists of the precautionary action zone (PAZ) and the urgent protective actions planning zone (UPZ) where arrangements have been made to take precautionary and urgent protective actions in the event of a nuclear or radiological emergency to avoid or minimize severe deterministic effects off the site and to avert doses off the site in accordance with international safety standards.

4.2 NPP siting criteria related with emergency preparedness

A nuclear power plant has several types of siting criteria (cf. Ch. 9.4), but related to emergency management are mainly population density & population centers, evacuation routes and possible external hazards. SMR plant providers are basically trying to have vastly more siting possibilities than available for large NPPs.

In Finland, STUK YVL guide A.2 ('Site for a nuclear facility', 12 pages, 70 numbered requirements) considers nuclear plant siting. However, in siting related questions, reference to several other YVL guides is made in requirement 205 of A.2:

- YVL A.1 Regulatory oversight of safety in the use of nuclear energy
- YVL A.3 Management system for a nuclear facility
- YVL A.7 Probabilistic risk assessment and risk management of a nuclear power plant
- YVL A.11 Security of a nuclear facility
- YVL B.1 Safety design of a nuclear power plant
- YVL B.7 Provisions for internal and external hazards at a nuclear facility
- YVL C.3 Limitation and monitoring of radioactive releases from a nuclear facility
- YVL C.4 Assessment of radiation doses to the public in the vicinity of a nuclear facility
- YVL C.5 Emergency arrangements of a nuclear power plant.

As already mentioned above in Ch. 4.1, in the US the siting criteria are different regulations from the EPZ. The INL (2010), willing to determine the EPZ for HTGR, state that 'the first step in considering sizing of the plume exposure EPZ and the ingestion pathway EPZ is to consider the regulatory requirements associated with siting and the related design considerations of NPPs of 10CFR100 (Reactor Siting Criteria) and 10CFR50 (Domestic Licensing of Production and Utilization Facilities). Two concepts that are defined in 10CFR100.3 and 10CFR50.2 are:

10 CFR 100.3: 'EA (Exclusion area) means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result.'

10 CFR 50.2: 'LPZ (Low population zone) means the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.'

For environmental reviews in the US, the NRC gave in October 2014 interim staff guidance COL/ESP-ISG-027 specifically titled 'Specific Environmental Guidance for Light Water SMR Reviews'.

In IAEA guidance, NS-G-3.2 (Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for NPPs) from 2002 (42 pages) considers the siting phase of an NPP project. The sizing of the EPZ, taking into account the surrounding population, is an important part of siting:

- Potential effects of NPP on the environment
- Surrounding area population distribution in site evaluation
- Main contents / atmosphere:
    - Source parameters (normal / accidental)
    - Necessary meteorological data
    - Instrumentation; Data collection, analysis & presentation
    - Modelling of atmospheric dispersion
- Main contents / hydrosphere:
    - Source parameters (normal / accidental)
    - Monitoring programme; surface / ground water
    - Uses of land & water, population distribution

Hadid Subki of IAEA (2013) presented results calculated by the ORNL for siting of both 1600 MWe and 350 MWe reactor (ORNL/TM-2011/157/R1) - see Figure 2 below.

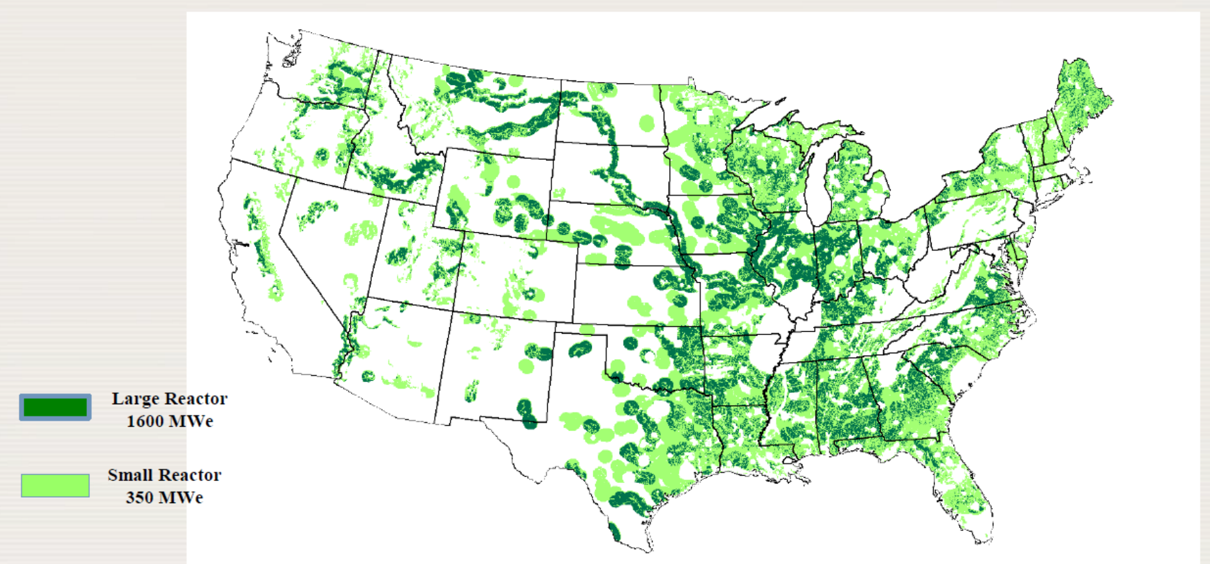


Figure 2. US siting options for small and large reactors, calculated by ORNL. Source of Figure: Subki, IAEA, 2013.

In the US, the DOE (Department of Energy) sponsored an ‘SMR Commercialization Workshop’ in June 2016. The participants included SMR vendors, utilities, national labs etc. The final report shows that the participants tried to ‘encourage DOE’ to support the industry in various topics, e.g. in siting flexibility:

‘Many of the unique characteristics of SMRs are expected to offer greater flexibility in selecting sites than with large LWRs. The incorporation of passive cooling systems and the smaller amount of nuclear fuel should facilitate NRC approval of smaller Emergency Planning Zones for SMRs. Some support reducing the EPZ to the site boundary. Reducing the size of the EPZ will allow the siting of SMRs closer to population centers while lowering the cost of emergency planning. DOE was encouraged to support industry’s efforts.’

The advantages of replacing aging coal plants with SMRs were discussed as follows:

‘The coal plant will have infrastructure – access to transmission, cooling water, administrative and other buildings – that can be used by an SMR. It also has skilled personnel that can be trained to operate and maintain the SMR. In addition, replacing one baseload resource, a coal plant, with another, an SMR, helps ensure continued stability of the electric grid.’

In Finland, to ensure CO2-free heat and electricity, similar high-level support for SMR ‘ecosystem’ would probably be needed.

Hidayatullah & Subki (2015) mention the following IAEA INPRO considerations regarding SMR installation site:

- Need for land use planning policies and controls over the population inhabiting the regions around the site
- Need for adequate emergency response strategies and planning (including evacuation locations)
- Site characterization infrastructure needed in place over the life of the facility
- Public engagement programs
- Access to adequate medical care, backup power, fuel and water
- Physical access to the site (whether by land, water or air)
- On-site security and emergency response capabilities

The IAEA SMR Regulator’s Forum DiD WG supports the recommendations of the 6th INPRO dialogue forum (2013), e.g. list of siting related concepts that require clarification. The DiD WG recommends to review e.g. IAEA, WENRA and USNRC requirements / recommendations, and possibly update them for SMR. They emphasize that the SMR design shall account for site-specific conditions to determine the maximum delay time by which offsite services need to be available.

4.3 EPR as the 5th level of Defence in Depth

EPR (Emergency Preparedness and Response) should be regarded as the 5th (and final) level of DiD. As the levels should be independent from each other, the EPR level should basically exist regardless of plant design. This is strongly emphasized in the report of the IAEA SMR Regulators’ Forum DiD WG:

‘In international and national standards and documents, the independence of the DiD levels is considered important for enhancing the effectiveness of DiD. Section 2.13 of SSR-2/1 (Rev. 1) states that the independent effectiveness of the different levels of defence is a necessary element of DiD. It helps to ensure that a single failure or combination of failures at one level does not jeopardize DiD at subsequent levels. The WENRA report, Safety of new NPP designs [A3], states that the levels of DiD shall be independent as far as is practicable. Lessons learned from the Fukushima Daiichi NPP accident have confirmed and reinforced the need for such a requirement. Therefore it should be applicable to SMRs as well. It could be investigated whether the SMR specific features, in particular the compact design of the modules and the multi modules design, may particularly challenge the independence of DiD levels.’

For DiD level 5, the DiD WG agreed with NEA statement that, no matter how much other levels may be strengthened, effective emergency arrangements and other responses are essential to cover the unexpected.

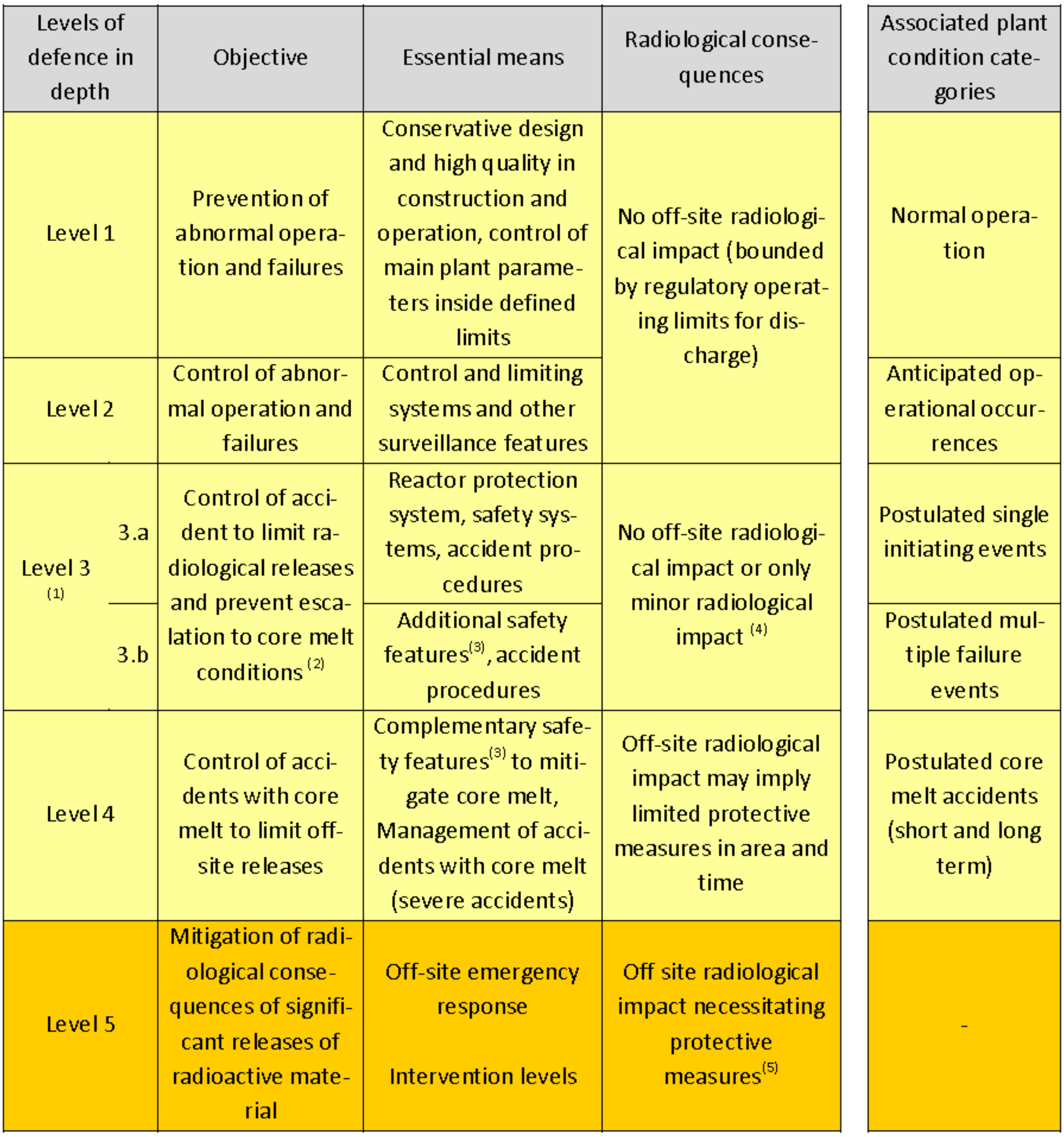
DiD in the Finnish regulatory guides on nuclear safety

The YVL guides, issued by STUK in Finland, set the requirements which must be fulfilled, or the applicant must prove that the safety level set forth is achieved. This is stated in the section 7 r(3) of the Nuclear Energy Act 990/1987 (changed 1 January, 2018 by 905/2017). The base of the YVL guides is the defence-in-depth principle (DiD). The DiD levels according to WENRA (defined in INSAG-10) are shown graphically in Fig. 3 and can be listed as follows:

1. Prevention of abnormal operation and failure
2. Control of abnormal operation and detection of failure
3. Control of accidents within the design basis
4. Control of severe conditions including prevention of accident progression and mitigation of the consequences of a severe accident
5. Mitigation of the radiological consequences of significant external releases of radioactive materials (‘What to do if all else failed?’)

The 5th level considers the situation where a radioactive release into the offsite environment already happened. Mitigation attempts in the hazardous situation are collectively called EPR (Emergency Preparedness and Response). The IAEA has defined safety requirements on EPR, given in GSR Part 7 (General Safety Requirements Part 7) of 2015.

The main topic of this report is the last DiD level, level 5. It is called the ‘mitigation of radiological consequences’ in the offsite environment of the nuclear plant. If level 5 is ever needed, it essentially means that the previous levels have already failed to accomplish their tasks, and a significant amount of radioactive material has been released out of the plant into the environment. It also basically means that something happened which was not considered at all or not thoroughly enough when the plant was designed and built.



Levels of defence in depth		Objective	Essential means	Radiological consequences	Associated plant condition categories
Level 1		Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation, control of main plant parameters inside defined limits	No off-site radiological impact (bounded by regulatory operating limits for discharge)	Normal operation
Level 2		Control of abnormal operation and failures	Control and limiting systems and other surveillance features		Anticipated operational occurrences
Level 3 (1)	3.a	Control of accident to limit radiological releases and prevent escalation to core melt conditions (2)	Reactor protection system, safety systems, accident procedures	No off-site radiological impact or only minor radiological impact (4)	Postulated single initiating events
	3.b		Additional safety features(3), accident procedures		Postulated multiple failure events
Level 4		Control of accidents with core melt to limit off-site releases	Complementary safety features(3) to mitigate core melt, Management of accidents with core melt (severe accidents)	Off-site radiological impact may imply limited protective measures in area and time	Postulated core melt accidents (short and long term)
Level 5		Mitigation of radiological consequences of significant releases of radioactive material	Off-site emergency response Intervention levels	Off site radiological impact necessitating protective measures(5)	-

Figure 3. Defence in depth levels according to WENRA [WENRA, 2013].

4.4 Protective measures in radiological emergencies

A short general introduction to protective radiological countermeasures in an offsite emergency is included here. The planning zones for emergencies, with some possible response actions, were explained in Ch. 4.1. In Finland, the main reference are the VAL Guides by STUK. The most important IAEA references are GSR Part 7 and EPR-NPP_PPA (2013).

IAEA GS-G-2.1 (Arrangements for Preparedness for a Nuclear or Radiological Emergency) from 2007 (159 pages) considers emergency arrangements, also other than purely radiological:

- Appropriate responses to a range of emergencies
- Background information on past experience

- Sources, types of emergency, public exposure, exposure pathways, health effects, countermeasures, threat assessment, threat categories, areas and zones
- Protective actions, public information, medical response, agricultural countermeasures, non-radiological consequences, response time objectives

Some important aspects of EPR that clearly had to be planned more thoroughly based on lessons learnt from past emergencies, particularly Fukushima 2011, include the following. Most of them may seem exaggerated when considering SMR plants, but some others, like public communication, very low probability events, and multiple units at the same site, may prove quite relevant.

- Justified protection strategy: Benefit has to outweigh the inevitable disadvantages of countermeasures.
- Optimized strategy: With finite resources, it is best to distribute them in a way that results in the best possible overall averted dose, provided that all individuals are sufficiently safe.
- Protection of emergency workers: It became evident in the Fukushima aftermath that the regulatory guidance for the dose limits & possible compensation etc. of emergency workers was not defined clearly enough.
- Vulnerable population groups: Evacuation is substantially more difficult for e.g. hospitals and elderly homes, and may cause more damage for the residents than the averted radiation.
- Waste generated during the emergency response: Particularly the amounts of various decontamination wastes may cause additional radiation protection problems.
- Simultaneous consideration of all hazards, not only radiological: In Fukushima, there was the overall destruction of infrastructure caused by the tsunami, to which the radioactive releases created an additional hazard. The general destruction hindered the performing of radiological response actions.
- Involvement of the medical community.

According to R. Bhattacharya (IAEA Expert mission), the objectives of emergency response can be listed as follows:

1. Gain control of the situation.
2. Mitigate the consequences.
3. Prevent deterministic health effects.
4. First aid and treatment of radiation injuries.
5. Reduce or prevent also adverse non-radiological effects.
6. Protect the environment and property.
7. Resume normal social and economic activity.

Some protective actions to be considered, together with their possible dose criteria (including exposure pathway & dose accumulation time period), are the following:

- Sheltering: 10 mSv of avertable dose in a period of no more than 2 days.
- Temporary evacuation: 50 mSv of avertable dose in a period of no more than 1 week.
- Iodine prophylaxis (blocking of the thyroid by pills of stable iodine): 100 mGy of avertable committed absorbed dose to the thyroid from radioiodine.
- Relocation of population (permanent resettlement): 30 mSv in 1 month or lifetime dose more than 1 Sv.

Some rather fundamental ‘cornerstones’ of radiation protection can be found in Publications of the ICRP (International Commission on Radiological Protection), notably the following two:

- ICRP 103 (334 pages, 2007): The 2007 recommendations of the ICRP
    - Deterministic / stochastic effects; embryo & fetus
    - Dose quantities
    - Optimization, reference levels, dose limits

  - Planned / emergency / existing exposure situations
  - Environment: animals & plants
- ICRP 109 (75 pages, 2008): Application of the Commission's Recommendations for the Protection of People in Emergency Exposure Situations
  - Protection of emergency workers
  - Projected / residual / averted dose
  - Justification & optimization
  - Protection strategy, termination of measures, permanent relocation
  - Contribution of different exposure pathways
  - Individual urgent protective measures

The IAEA has applied ICRP recommendations regarding radiation protection for the practical purposes within nuclear energy use. The most important IAEA requirements / guides for nuclear offsite emergency preparedness are GSR (General Safety Requirements) Part 7 (IAEA 2015) and EPR-NPP-PPA (IAEA 2013). Regardless of the size of a nuclear plant (including also SMR), the potentially needed protective countermeasure actions depend on carefully set dose thresholds, but also implementation considerations, like time windows (the right time to perform the action) and existing infrastructure (practical means to perform the action).

The EPR-NPP_PPA states that the goals of the protective actions and other response actions are to:
- Prevent the occurrence of severe deterministic effects; and
- Keep the doses below the generic criteria at which protective actions and other response actions are justified to reduce the risk of stochastic effects.

For emergency preparedness planning, the predicted dose levels in SMR offsite environment should be extensively compared with STUK / IAEA criteria for protective radiological countermeasures, like sheltering indoors, iodine pills or evacuation. There are two essential considerations: is the criterion exceeded, and if it is, will it be practically possible to perform the countermeasure action without causing more harm than benefit?

Only then is it possible to give justified recommendations of EPZ size, based on expectedly needed protective measures, in some cases possibly considering appropriate scaling down from large reactor EPZ approach.

An adquate offsite 'all hazards plan' should include defining emergency action levels (EALs), emergency drills and training, protective action strategies, and a modern public alert system. A complete protection strategy should include the following definitions:

- EAL: emergency action levels (threshold for a plant condition to decide the emergency class; see the EPR-NPP_PPA (IAEA 2013) Ch. 3 for the recommended 'Emergency classification system'
- OIL: operational intervention levels / limits (field and possibly laboratory measurements of e.g. deposition Bq/m2 or water contamination Bq/kg)
- Emergency planning zones and distances
  - PAZ (precautionary action zone), e.g. approximately 5 km
  - UPZ (urgent protective action zone), e.g. approximately 20 km
  - EPD (extended planning distance), e.g. 100 km
  - ICPD (ingestion and commodities planning distance), e.g. 300 km
- Response actions, for each EAL and OIL
- GC (generic criteria), defined by projected/received doses (see GSR Part 7, Appendix II: Generic criteria for use in EPR)

Consideration of all hazards (possibly also other than the radiological hazard may contribute to the emergency), together with all their consequences, is important. Also the dose-averting countermeasures have both radiological and non-radiological (societal, economic) consequences.

The recommendations in national guidelines, like the VAL guides of STUK (Finland) or NRC regulatory guides & EPA (US environmental protection agency) PAGs (protective action guidelines) are generally quite compliant with IAEA GSR Part 7. The STUK emergency guidelines (VAL, e.g. STUK 2012) are the following:

- VAL-1: Protective measures in early phase of radiation emergency (Latest version, of 100 pages, published in September 2020)
  - Staying indoors, iodine pills, evacuation, etc.
- VAL-2: Protective measures in late phase of radiation emergency
  - Staying indoors, evacuation, relocation of population, decontamination, etc.
  - In VAL-2, chapter 4.5 (Protection of people working or staying outdoors in contaminated areas) covers radiation protection of workers.
- VAL-3: External radiation monitoring guidelines for rescue staff
- VAL-4: Requirements for portable radiation detectors

The phases of a nuclear / radiological emergency can be explained as follows:

- Early phase
  - Before & during radioactive release
  - Radiation level in the environment is increasing
  - Ends when plume has passed & no further releases
- Intermediate phase
  - Lasting from a few days to a few years
  - Radioactive material mainly on ground (decay, migration)
  - Re-define the protective measures
- Recovery phase
  - ’Back to new normal’, with possible long-term actions
- Further references: Protective Measures in Early and Intermediate Phase of a Nuclear or Radiological Emergency, Nordic Guidelines and Recommendations (2014)

Some examples of intervention levels for countermeasures in the VAL guides:

- Sheltering indoors, if dose > 10 mSv / 2 d, or OIL: dose rate > 100 µSv/h
- Iodine prophylaxis, if dose rate > 100 µSv/h, or concentration > 10 kBq/m3 for 2 days
- Evacuation, if expected effective dose > 20 mSv in 1 week
- Protecting food production OILs: external dose rate > 1 µSv/h, or limits of air concentration are exceeded

Some of the most important protective measures can be characterized as follows:

Sheltering indoors:
- If dose > 10 mSv / 2 d, or OIL: dose rate > 100 µSv/h
- Reduce both inhalation and external exposure
  - Try to make the building as airtight as possible
  - Best shielding in lower & central parts of building
- Results affected by building type & air exchange rate
- May be complete or partial (part-time / children etc.)
- Usually lasting from a few hours to 1 day
- After cloud passage: ventilation, possible decontamination

Iodine prophylaxis:
- Prevent accumulation of radio-iodine in the thyroid
- Thyroid dose may cause thyroid cancer
- Protecting only the tyroid, only from iodine nuclides

- Most crucial for children & pregnant women
- Only after orders issued by the safety authority
- No going to pharmacy after indoors sheltering orderd!
- Pills of KI (stable kalium iodide) taken 1-6 h before exposure, protection for 1 day
- Dose rate > 100 μSv/h, or c_I > 10 kBq/m3 for 2 days

Evacuation:

- If expected effective dose > 20 mSv in 1 week
- As early as possible: possibly before arrival of the plume
- Pitfalls: Panic among the public, exposure en route
- ETE = Evacuation Time Estimate (possibly from simulation model)
- Present-day advanced communication systems might decrease the time needed

Protection of emergency workers:

- OIL: external dose rate 10 μSv/h – 1 mSv/h
- Protective clothing, respiratory protection
- Iodine tablets: In predefined risk facilities, or after a general recommendation of iodine prophylaxis
- Working time, locations and measured dose rates recorded
- Restrict working time in order not to exceed total of 50 mSv
- Higher dose rates: corresponding working time restrictions
- Limit up to 500 mSv may be accepted in order to:
    - Save lives or prevent severe health effects
    - Prevent catastrophic conditions
- Workers must be clearly informed about risks & protection
- Radiological monitoring, medical surveillance

Protection of food and other goods:

- Start protecting food production as soon as possible
- OILs: external dose rate > 1 μSv/h, or limits of air concentration are exceeded
- Even with external dose rates very close to background, radionuclides may accumulate into foodstuff dangerously
- Radioactive iodine (half-life 8 d) reaches milk very fast
- In Chernobyl 1986, the Soviet Union failed to protect milk
- Clean feed & drinking water to livestock
- OILs for raw materials, products & factories: external dose rate > 10 μSv/h, or limits of air concentration are exceeded
- Factories etc: stop ventilation, stop production
- Historic example: contaminated packaging destroyed films

Protection of the public in the intermediate phase:

- Sheltering indoors (for external dose rate; maybe partial)
- Restrictions to enter contaminated area
    - External dose rate > 100 μSv/h: only necessary entries
- Evacuation & relocation
    - Consider limits & difficulty of moving people
    - Relocation, if expected 10 mSv during 1st month
- Reducing exposure of inhabitants
    - Cleaning, washing (e.g. hands), outdoor clothes, air filters
- Protection of workers (necessary work)
    - Decontamination, repairs, surveys, waste disposal, health
- Members of public working in the area
    - Emplyer’s responsibility, limits as members of the public
- Decontamination & monitoring
    - Suitable local stations; washing, disposal of clothes

In the US, it is the responsibility of the EPA (Environmental Protection Agency) to publish the Protective Action Guides (PAGs) which help responders to plan for radiation emergencies. The latest PAG manual (EPA, 2017; 112 pages) has the following chapters:

- Ch. 2: Early phase protective action guides (p. 13-32)
- Ch. 3: Emergency worker protection (p. 33-39)
- Ch. 4: Intermediate phase protective action guides (p. 40-68)
- Ch. 5: Planning guidance for the late phase (p. 69-86)

INL (Idaho National Laboratory) have studied the problem of determining an adequately sized EPZ and the related emergency planning for a 200 MW HTGR (high temperature gas-cooled reactor) plant (INL, 2010). Their extensive 65-pages study considers many aspects of protective countermeasures like graded approach, local emergency response capabilities and interfaces with coordinating agencies. It is not possible to describe the details here, but the reader may refer to INL (2010) for e.g. the following topics:

- EPZ Sizing
- Identifying the Applicable Source Term
- Calculation Methodology for Offsite Dose Consequences
- Factors Other Than Offsite Dose Consequences that Influence EPZ Size
- Regulatory Basis for EPZ Sizing
- Graded Approach to Emergency Plans for the HTGR
- Emergency Planning Requirements with Corresponding EPZ Sizing
- Emergency Action Levels / Initiating Conditions for LWRs
- Relationship to Local Emergency Response Needs and Capabilities
- Assessment of Interfaces with, and Requirements of, Coordinating Agencies
- Licensing Document Structure and Regulatory Impacts
- Proposed Changes to Regulatory Guidance Documents or Requirements

In Finland, an example of the planning of an offsite emergency services department can be found e.g. in www.iupela.fi (Emergency services department Itä-Uusimaa, for Loviisa NPP; see: Itä-Uudenmaan pelastuslaitos, 2013):
LOVIISAN_VOIMALAITOS_PELASTUSSUUNNITELMA_15_3_2013_JULKINEN.pdf

After reviewing the NEI SMR EPZ methodology (2013), the NRC included in their feedback questions also an inquiry about the modeling of evacuation & other response actions: 'Are emergency response actions, such as evacuation modeled for all three criteria, using MACCS? If evacuation is modeled in the analyses, provide a discussion on why this is appropriate.'

Evacuation of large numbers of people is by no means straightforward and poses many other threats than radiation, e.g. traffic jams, psychological panic, and the stress of having to leave one's homes. An evacuation in the US caused by a storm was described in Scientific American (12 September 2018, 'How to evacuate cities before hurricanes') with the following words:
'*When cities near the coast like Houston face severe storms, evacuations seem the obvious way to protect people. But moving millions of people carries its own dangers. When Rita took aim at our area in 2005, officials told everyone to leave. Giant traffic jams turned Interstates 45 and 10 and U.S. Route 59 into parking lots as people at low risk fled, blocking escape routes for individuals who needed them most—residents directly in the path of high winds, heavy rain and storm surge. A few died on the road in the tremendous heat. A bus evacuating residents from a nursing home caught fire, igniting an oxygen tank and killing 23 onboard. So when Hurricane Harvey bore down on Houston last August, Mayor Sylvester Turner refused to evacuate. You literally cannot put 6.5 million people on the road, he said at the time; If you think the situation right now is bad, you give an order to evacuate, you are creating a nightmare.*'

4.5 Offsite dose limits and EPZ size required in Finland

EPZ size regulation in Finland is currently more straightly prescriptive than performance oriented (or, consequence oriented). The values 5 km (PAZ) and 20 km (UPZ) are given for nuclear power plants in the STUK YVL guides and even at higher level. The 5 km zone can be interpreted as the area that can be effectively evacuated. (Note however: The zones were not required for the VTT research reactor at Otaniemi, Espoo, under decommissioning as of 2021.)

The basis for the zones in Finland was reviewed in 1984 in a STUK statement to the Ministry of Interior. According to information gathered in IAEA TECDOC 1652 (IAEA 2010; report of a CRP project), 'State-of-the-art severe accident management and source term were considered. Description of possible accident scenarios and times, radioactive substances behavior at the plant and release phenomena were drawn into a perspective of possible protective actions.'

Historically, the YVL requirements referred to Government Decrees (VnA):

- 411: Government Decree (717/2013) stipulates that a **precautionary action zone** shall surround the site area and extend to a distance of approximately 5 kilometres from the plant, and that land use restrictions are in force in this area. The precautionary action zone shall include in their entirety any villages and settlements that are located inside the area. The following aspects supplement requirement 402…
- 413: In accordance with the Government Decree (716/2013), the facility shall be surrounded by an **emergency planning zone** extending to about 20 kilometres from the plant; the zone shall be covered by a detailed external rescue plan for the protection of the public drawn up by authorities. The precautionary action zone shall be part of the emergency planning zone.

On 1 January 2016, Government Decrees (VnA, Valtioneuvoston asetus) were replaced by STUK regulations (STUKin määräykset), like

- 42423 STUK Y/1/2016: STUK regulation on nuclear power plant safety (previously Government Decree 717/2013)
- 42424 STUK Y/2/2016: STUK regulation on emergency arrangements at an NPP:
  Section 2 defines PAZ (precautionary action zone, sometimes also known as protection zone, or 'suojavyöhyke') as approximately 5 km from the NPP and EPZ (emergency planning zone, 'varautumisalue') as appr. 20 km.

The EPZ sizes of Y/2/2016 are stated once again at a lower level in the YVL Guide A.2 (previously YVL 1.10), in requirements 411-413:

- 411: Land use restrictions in the 5 km PAZ
  - No schools, hospitals, care facilities, shops, or significant places of employment or accommodation (except the NPP); no densely populated areas
  - No socially significant functions that could be affected by an NPP accident
  - Number of permanent inhabitants, recreational housing and activities limited (to appr. 200) to allow effective evacuation
  - Permanent and leisure-time population must not increase substantially during NPP construction and operation
- 412: Reference to licensee duties within PAZ
- 413: The EPZ (about 20 km) must have detailed external rescue plan for the protection of the public.

Within 1 km from an NPP, there should be no permanent dwellings. An example of the PAZ and EPZ zones in Finland is shown in Fig. 4 using Loviisa NPP surroundings as an example.

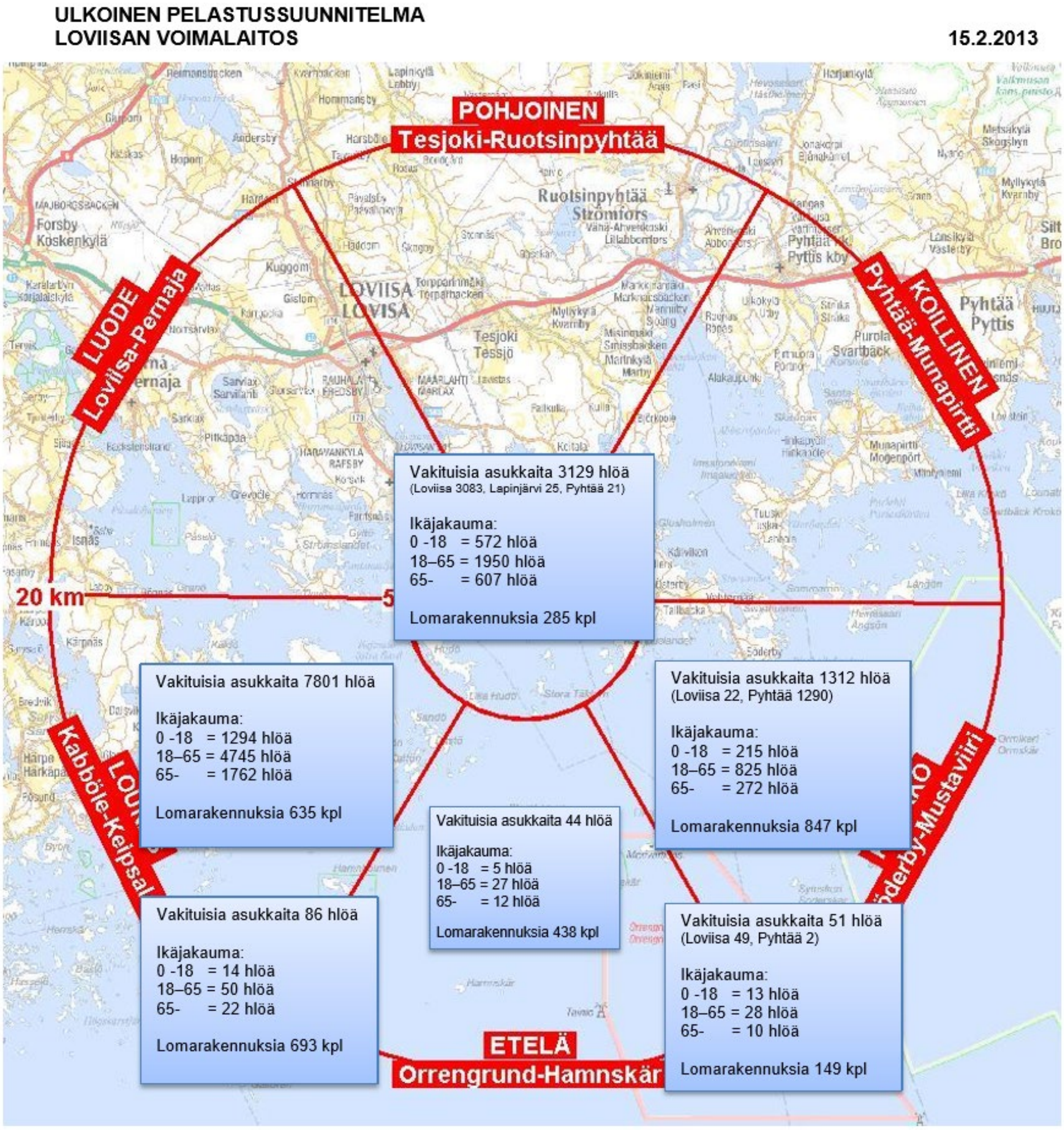


Figure 4. Public rescue plan for Loviisa NPP: Map showing PAZ (5 km), EPZ (20 km) and the 6 emergency preparedness sectors. Indicated are the numbers of permanent inhabitants, their age distributions, and the number of recreational buildings (summer cottages). Source of Figure: Itä-Uudenmaan pelastuslaitos, Loviisan laitoksen julkinen pelastussuunnitelma, 2013.

In the Graded Approach Working Group (GA WG) report of IAEA SMR Regulators' Forum, it is mentioned about Finland (page 39) that the size of the EPZ in Finland is site-specific. This should probably be interpreted in the way that the licensee has to show for his selected site, using site-specific data, that a larger area is not needed to implement protection of the public. By e.g. VTT research activities, most recently in the CASA project of SAFIR2018, the present zone sizes in Finland have been checked many times against calculated offsite doses. It was concluded that if the radioactive release is limited to the 100 TBq allowed for severe accidents, there will be no need of urgent protective actions outside the 20 km zone.

Limits for radiation exposure of members of the public (population) are given in STUK Regulation Y/1/2016 on the Safety of a nuclear power plant, Sections 8-10, for different plant conditions:

- The differences reflect the frequency of the conditions:
    - 0.1 mSv/a for normal operation
    - 0.1 mSv/a for AOO (anticipated operational occurrences)
    - Accidents
        - 0.1 mSv/a for Class 1 DBA (Design Basis Accident)
        - 5 mSv/a for Class 2 (p < 1/1000 reactor years) DBA
        - 20 mSv/a for DEC (Design Extension Condition; SA)
- Severe accidents (SA), with at least partial core melt:
    - ’SA release must not necessitate sheltering or long-term land restrictions over wide areas.’
    - Cs-137 release (largely determining the long-term effects) must not exceed 100 TBq
    - Large early release (LER) must be ’very improbable’ (no exact LERF)

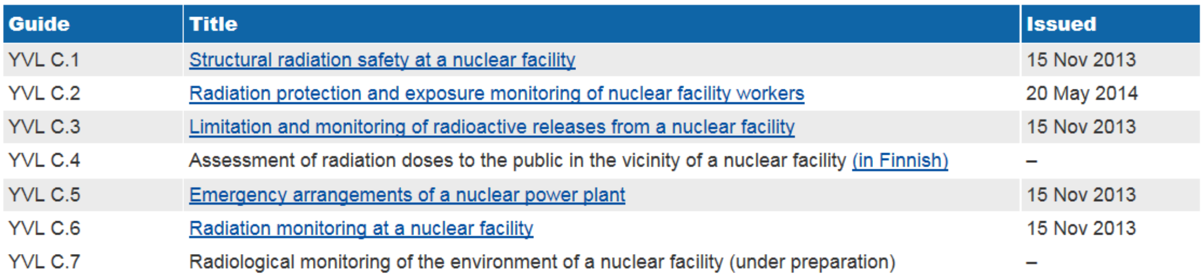
The exact wording of what is required in the case of a radioactive release caused by a severe accident has changed somewhat over the years in Finland. Government decision 395/1991 (refered to by YVL 7.1 / 22 March 2006) had the following in 12 §:

’The limiting release for a radioactive release caused by a severe reactor accident is such that it will not cause acute health effects to the offsite population and no long-term restrictions on the use of large areas of land or water. To fulfill the long-term requirement, the limit for atmospheric release of Cs-137 is set at 100 TBq.’

Later, Government Decree 717/2013 (Ch. 3, Section 10) does not mention health effects:

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

At present, STUK YVL Guides in Group C consider radiation safety of a nuclear facility and environment:



Guide	Title	Issued
YVL C.1	Structural radiation safety at a nuclear facility	15 Nov 2013
YVL C.2	Radiation protection and exposure monitoring of nuclear facility workers	20 May 2014
YVL C.3	Limitation and monitoring of radioactive releases from a nuclear facility	15 Nov 2013
YVL C.4	Assessment of radiation doses to the public in the vicinity of a nuclear facility (in Finnish)	–
YVL C.5	Emergency arrangements of a nuclear power plant	15 Nov 2013
YVL C.6	Radiation monitoring at a nuclear facility	15 Nov 2013
YVL C.7	Radiological monitoring of the environment of a nuclear facility (under preparation)	–

Assessment of doses to the public may be needed e.g. for the following purposes:

- For safety analyses of PSAR / FSAR reports
- During operation of the plant (calculational assessment of doses)
- For a full-scope PSA study
- In radiological emergency situations (for dose rate predictions)

The YVL Guide C.4 considers (quite detailed) requirements for assessment of radiation doses to the public in the vicinity of a nuclear facility, with the following main contents:

- Dispersion calculations of radioactive releases
- Assessment of doses to the population in the environment
- Requirements for meteorological measurements

Cf. also YVL guide B.3 (Deterministic safety analysis), Ch. 5: Release and dose analyses.

The STUK emergency response guidelines (VAL-ohjeet) were considered in Ch. 4.4 (Protective countermeasures) of this report. They can be found from the following internet address: https://www.stuklex.fi/fi/ls#valmiusohjeet

4.6 Recent developments in Finland (MEAE and STUK)

The SMR EPZ problem is partly related with the coming reform of nuclear legislation in Finland. The existing legislative framework seems too complex for new operating models & technologies. Present licensing not suitable for SMR. At the same time, climate change has become a key theme. It seems that SMR-specific requirements will be accounted for in the reform of nuclear legislation, which will also include the regulation of spent fuel disposal.

Currently Finland has a three-step licensing scheme for nuclear newbuild:

1. Decision in Principle (DiP) - January 2002 for Olkiluoto 3
    - Presently, the DiP is not needed if $P_{th} < 50$ MW
2. Construction License - February 2005 for OL3
3. Operating License - March 2019 for OL3

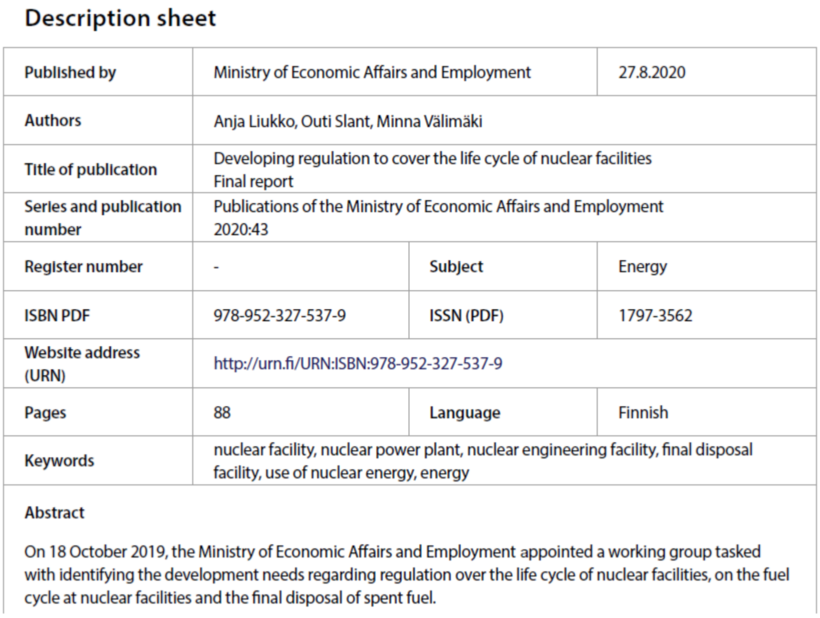
To facilitate possible future licensing of SMR plants (among many other reasons considering the whole nuclear life cycle), MEAE (Finnish Ministry of Economic Affairs and Employment) and STUK (Radiation and Nuclear Safety Authority in Finland) have initiated a major reform of the nuclear regulatory system in Finland. According to information from STUK about recent regulatory developments in Finland, an extensive update is expected to begin, considering:

- Nuclear Energy Act & Decree (MEAE)
- The YVL & VAL Regulatory guides (STUK)

A working group on the reform was set by MEAE in October 2019 and gave their final report on 27 Aug 2020: Regarding possible SMR plants, there might not be a separate 'line' of licensing, but rather a new process suitable for both big plants & SMR (considering both technical requirements & the process itself). In the 2020 final report (TEM 2020:43), the group writes 'To reduce uncertainties and to speed up nuclear projects, there could be an advance approval of plant concept and site. Approved items would not be reconsidered in the construction license phase, provided that they have remained as approved. There could be some flexibility, case by case, in the required extent and depth of the advance approval.' Such a preliminary assessment of plant design would resemble the DCA (Design Certification Application) and ESP (Early Site Permit) in the US, or the 'pre-licensing Vendor Design Review' (VDR) in Canada. There are still many open questions about the details of the future process in Finland:

- Who has the authority to issue the approval? What is the cost for the applicant?
- Who is eligible to apply for an advance approval?
- Other preconditions for submitting an application?

- What is exact extent of the applications / approvals?
- How long are the approvals valid?
- What happens if there are changes to the design or to the site after the approvals?
- What happens to the approval if there are any changes in legislation or other regulation afterwards?
- Relation of site permit with urban planning and EIA (Environmental Impact Assessment)



Description sheet

Published by	Ministry of Economic Affairs and Employment		27.8.2020
Authors	Anja Liukko, Outi Slant, Minna Välimäki		
Title of publication	Developing regulation to cover the life cycle of nuclear facilities Final report		
Series and publication number	Publications of the Ministry of Economic Affairs and Employment 2020:43		
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Pages	88	**Language**	Finnish
Keywords	nuclear facility, nuclear power plant, nuclear engineering facility, final disposal facility, use of nuclear energy, energy		

Abstract

On 18 October 2019, the Ministry of Economic Affairs and Employment appointed a working group tasked with identifying the development needs regarding regulation over the life cycle of nuclear facilities, on the fuel cycle at nuclear facilities and the final disposal of spent fuel.

MEAE (27 August 2020): 'Developing regulation to cover the life cycle of nuclear facilities'

Considering the separate site assessment before construction licence (cf. 'ESP'), emergency planning zone (EPZ) sizing is clearly a central question, to be studied further in Finland

In 2020, STUK published their outlook for future regulatory control as 'Preconditions for the safe use of small modular reactors' (STUK 2020). Regarding the SMR EPZ problem, the main messages are that the safety of new reactor types must be demonstrated reliably and that the purpose of precautionary action zones and emergency planning zones is to protect people, so the size of the zones is to be considered according to need:

- The present EPZ size is based on possible accidents in current, large NPPs.
- It is necessary to assess the size with regard to the consequences of possible accidents in SMRs (cf. 'performance-based').
- Precautionary action zones and emergency planning zones are a key issue, especially in the utilization of SMRs in district heating, where plants must be located relatively close to the customers.
- To cite directly what STUK says about nuclear district heating, '*In the case of district heating production, a plant producing heat must be located relatively close to habitation. The size of the precautionary action zone (PAZ) and the emergency planning zone (EPZ) must be considered according to need on the basis of the risk caused to the surroundings of the plant.*' Definitely here the keywords are need and risk.
- Most probably, international guidance, like by IAEA, would be very helpful in developing national guides.

From VTT participation in 'Offsite dispersion and dose assessment expert meetings' at STUK (2021) it can be mentioned that STUK has also performed some dose assessments using publicly available SMR core radioactive inventory information, but those reports were not made public.

4.7 Emergency planning zones and distances in Sweden

EPZ size regulation in Sweden is currently expected to change, with implementation in contingency planning by 1 July 2022. Four years after Fukushima accident, in 2015, the government asked the SSM (Swedish radiation safety authority) to review the sizing of emergency planning zones and distances. In the 2017 report (SSM 2017), comprising with appendices about 300 pages with details about the assumptions, calculation methods and criteria, SSM proposed to change the distances to PAZ 5 km, UPZ 25 km (previously 12…15 km) and EPD (Extended Planning Distance) 100 km for large NPPs. For other nuclear plants (fuel fabrication, SNF interim storage) in Sweden, the distances ranged from 700 m to 2000 m. As only existing plants were considered, there is no direct result for SMRs. However, the used methodology shows that there is no need to resort to fixed EPZ size for all kinds of plants. The methodology seems basically rigorous, but is not extensively based on PRA-calculated frequencies of accident cases.

The report SSM 2017:27e, Review of Swedish emergency planning zones and distances, uses assessment methodology that can be briefly described with the following steps; cf. also Table 1 of SSM (2017).

- Postulated events at the NPPs (SSM App. 3; cf. Ch. 7 of this report)
- Source term: nuclides, activities, duration, height etc. (SSM App. 3; cf. Ch. 8 below)
- Geographical domain (SSM App. 2; cf. Ch. 9.4 of this report)
- Meteorological data / scenarios (SSM App. 2; cf. Ch. 9.3 of this report)
- Atmospheric dispersion modelling, RIMPUFF model (SSM App. 2; cf. Ch. 9 below)
- Shielding factors for cloudshine and groundshine (SSM App. 1; cf. Ch. 10 below)
- Dose calculations (SSM App. 2; cf. Ch. 10 of this report)
- Dose criteria for protective actions (SSM App. 1; cf. Ch. 4 of this report)
- Sensitivity studies (SSM App. 3; cf. Ch. 11 of this report)
- Frequency distributions (SSM App. 2; cf. Ch. 11 of this report)
- Maximum distances (SSM App. 2; cf. Ch. 11 of this report)
- Final decision method (SSM App. 2; cf. Ch. 11 of this report)

4.8 EPZ regulations in other nuclear energy countries

In spite of many attempts of international harmonization, over many decades, the regulations on EPZ and emergency management in general have big country-specific differences. Some sources of international references can be listed as follows:

- ICRP (International Commission on Radiological Protection)
- IAEA: Safety Guides, SMR Regulator's Forum
- NEA (OECD nuclear energy agency) recommendations
- EC (European Commission) Basic safety standards (2013/59)
- EC (European Commission) directive on nuclear safety (2009/71 and 2014/87)
- NERIS platform (European Nuclear and radiological Emergency management and Rehabilitation strategies Information web Site)
- WENRA (Western European Nuclear Regulators Association) safety objectives (Reactor Harmonization Working Group, RHWG 2013)
- EUR (European Utility Requirements)

The IAEA SMR Regulators' Forum EPZ WG surveyed some member states for their current approaches of determining EPZ size. The EPZ distances (Table 2) and concise information on how the offsite consequences are verified were found to be as below. The EPZ WG final report (2018) contains sections of each country's specific information at the end.

Table 2. Emergency planning zones and distances in 6 IAEA member states, by the survey of the EPZ WG. For explanations, see numbered list below.

	EPZ		EPD			Verification of Source Term/Offsite Consequences [see itemized list below]	
Country	PAZ	UAZ	EPD	ICPD			
Canada	Not pre-determined					(2)	
China	7-10 km		30-50 km			(5)	
France	20 km		20 km			(3)	
Korea							
Russian Federation	<25 km		<100 km			(4)	
USA	16 km (10 miles)		80 km(50 miles)			(1)	

1. USA: The use of approved codes and methodology, the regulators require the input and output files for the verification of the source terms and offsite consequences. If the applicant uses a method or code other than an approved, the applicant must supply the input and output files and the source codes for the computer modelling that support the analysis and determination of source terms and offsite consequences with respect to the specific designs are part of an application.
2. Canada: The applicant needs to provide all relevant information for the offsite authorities assess or make an informed decision on the EPZ, such as the source term and accident sequences. The calculation is not required. Note that the EPZ size is not pre-determined (cf. Ch. 5.3). PSA, including level 3, and DSA (deterministic safety analysis) are both used.
3. France: The applicant needs to provide all relevant information for the offsite authorities assess or make an informed decision on the EPZ, such as the source term and accident sequences. The calculations need to be included in the safety case. Post-accident zoning is defined with ZPP (public protection zone) and ZST (heightened territorial surveillance zone).
4. Russia: Offsite consequences are verified by using nuclear regulator guidance and safety review.
5. China: The applicant needs to provide all relevant information for determining the EPZ, such as source term and accident sequences. All above should be in accordance with nuclear safety regulations. The Chinese terminology for siting has Exclusion Area Boundary (EAB), Planning Restricted Area (PRA) and the EPZ. In the country-specific section, plume EPZ & ingestion EPZ and siting regulations are explained.

Russia

In Russia, the EPZ size is set by Rostechnadzor (the nuclear regulator) & EMERCOM (ministry of emergency situations). The EPZ sizes for commercial reactors according to the EMERCOM document ‘Standard contents of off-site protection plan’ (2006) are shown in Table 3 below.

Table 3. Commercial reactor EPZ sizes by the Russian EMERCOM. Source of Table: IAEA SMR Regulators' Forum (2018).

Thermal power, MWt	Radius of precautionary protective actions planning zone, km	Radius of urgent protective actions planning zone, km		Radius of intermediate and longer term protective actions planning zone, km	
		inner	outer	inner	outer
> 1000	5	5	25	25	100
100 - 1000	3	3	25	25	100
10 - 100	n.a.	0	5	5	50
2 - 10	n.a.	0	0,5	0,5	5

Ramana et al. (2013) provide some SMR-specific information about Russia: 'The siting of nuclear reactors in Russia is governed by NP 031-01 on the Siting of nuclear power plants that states that the EPZ radius shall not be more than 25 km (5 km in the case of reactors used for heating) from the site boundary of the reactor. However, for the SMRs under development, the EPZ radii are much smaller: 1 km for the KLT-40S, VBER-300 and ABV, while the EPZ radius for the SVBR-100 has not yet been specified.'

South Korea

According to KAERI (2018), there is a new strategy of emergency management in South Korea, with the goals of preventing the occurrence of deterministic health effects and reducing the occurrence of stochastic health effects in emergency situations. In the new strategy, PAZ should be 3-5 km and UPZ 20-30 km. In the case of a research reactor, the EPZ was set at 1500 m, after extension request of local government.

Mancini et al. (2014) suggest a PRA-based method for EPZ determination for SMR. They have surveyed the current EPZ sizes in four countries (Table 4 below).

Table 4. Current EPZ size in USA, France, Spain and Japan. Source of Table: Mancini et al. (2014).

USA	10 miles	Plume exposure pathway	Exclusion area	Total radiation dose to whole body in 2 h>25 rem Total radiation dose to the thyroid from iodine exposure in 2 h>300 rem
	50 miles	Ingestion exposure pathway	Low population zone	Total radiation dose to whole body during the entire period of passage>25 rem Total radiation dose to the thyroid from iodine exposure during the entire period of passage>300 rem
France	5 km		Evacuation pre-planned	
	10 km		Sheltering pre-planned Stable iodine tablets distributed	
	>10 km		Possible extension of protective actions	
Spain	10 km		Sheltering, evacuation and stable iodine intake in the preference sector	
	30 km		Food restrictions	
Japan	8–10 km		Lower limit of radiation exposure between $D < 10$ mSv whole body $D < 100$ mSv thyroid	

UK

The EPZs in the UK are, according to crarisk.com, currently based on deterministic criteria. The Office for Nuclear Regulation (ONR) sets the detailed area, e.g. 3-4 km for Sizewell B. Currently the ONR is revising the principles for determining detailed EPZs (‘Revised requirements for radiological protection’).

India

About India, Ramana et al. (2013) say that the current EPZ size is 10 miles for any reactor power level according to the AERB (Atomic Energy Regulatory Board), to be replaced by NSRA (Nuclear Safety Regulatory Authority).

USA

INL (2010), page 30, gives a short account on the US regulatory basis for EPZ sizing and emergency planning requirements. From the short USA-specific section of the IAEA SMR Regulators’ Forum EPZ WG, the main points can be summarized as follows:

- USA regulations dictate fixed distances around large LWR: 10 miles (plume exposure, PEP) and 50 miles (ingestion, IEP).
- The same distances are given in 10CFR (Energy) and 44CFR (Emergency management and assistance).
- NRC regulates the nuclear licensees whereas FEMA (Federal emergency management agency) inspects/evaluates EPR for offsite communities.
- However, the current regulations are for LWR > 300 MWe.
- NRC is currently developing EP for SMR, advanced reactors & isotope production / utilization (rule identification number 3150-AJ68). The NRC (2017) draft regulatory basis for SMR EP rulemaking describes the existing regulatory framework & NUREG-0396 methodology, and regulatory issues caused by SMRs differing substantially from the existing large LWR fleet.
- In current NRC practice, SMR should request for an exemption, providing the necessary information and analysis.

The TECDOC 1652 document of IAEA CRP I25001 (IAEA 2010, Annex I) describes the history of the present EPZ zones in the US as based on the key document NUREG-0396 (NRC 1978), produced by a Task Force of the NRC and EPA set in 1976. The major recommendations were:

1. Enlarged spectrum of considered accidents, from DBA even up to total core melt with degradation of containment boundary
2. Use of PAG (Protective Action Guidelines) limits
3. Separate guidance for two exposure pathways (plume PEP and ingestion IEP)

Several criteria for considering or excluding accidents were discussed:

1. Risk: Theoretically, the most beautiful choice would be the simple mathematical concept of risk as ‘probability x consequence’. This would leave out accident sequences with very low probability or minor consequence, or both.
2. Probability: Exclude very low-probability accidents by their probability alone.
3. Cost-effectiveness: Cost of emergency planning vs. averted consequences.
4. Consequences: Exclude only those accidents whose radiological consequences are small enough.

In the 0396, consequences (#4 above) were chosen as the rationale basis. However, very unprobable events were left out by probability considerations. As the main result the concept of the Emergency Planning Zone (EPZ) was developed, as the area where planning for protective actions is needed. The EPZ would consist of the PEP area with 10 miles radius and IEP area with 50 miles radius. These radii were based on the following dose criteria:

1. DBA or 'less severe Class 9' accident consequences would not exceed PAG levels outside the EPZ.
2. In case of 'more severe Class 9' accidents, EPZ size is sufficient for substantial reduction in early severe health effects (radiation injury or death).

To have the EPZ site-independent, the radii were also chosen to meet the dose criteria for any given NPP. (Obviously, the 0396 Task Force was thinking of relatively large NPPs of approximately similar size.)

The SMR vendor NuScale gives an easily understandable description of the US EPZ distances for current large NPPs: '*In practice there are two EPZs surrounding the plant site. The first, called a Plume Exposure Pathway (PEP), is traditionally at a 10 miles radius for conventional nuclear plants, and is designed to avoid or reduce the dose from potential exposure of radioactive materials from the plant. The second, called the Ingestion Exposure Pathway (IEP), is about 50 miles in radius for conventional nuclear plants, to avoid or reduce the dose from potential ingestion of food contaminated by radioactive materials. For both zones specific emergency procedures are in effect.*'

Ingersoll (2009) describes the US situation: 'Despite the excellent safety record of the current fleet of U.S. nuclear plants, nuclear reactors are still considered a high-risk technology that should be centralized in remote locations. This notion is exacerbated by NRC regulations that require large (10 mile) emergency planning zones around nuclear plants. This represents only one of many regulatory biases toward large plants that must be changed if many of the benefits of SMRs are to be realized. The NRC is currently working to develop a new licensing framework that is more neutral to non-LWR technology.'

Many authors point out that the current zones in the US were not the result of rigorous and definitive scientific analysis (as should be and could be, as suggested in this report), but rather the result of subjective qualitative judgements:

- Talabi & Fischbeck (2016) describe the 'pseudo-quantitative approach used in 1978 assessment': The selection of a 10-mile radius was determined based on an examination of the curves that showed the dose 'decrease sharply within 10 miles and to decrease slowly at greater distances'. Hence, the determination of the 10-mile radius was subjectively determined based on subject-matter experts' review of the data. The lack of a quantitatively determined basis for the 10-mile radius makes it challenging to draw inferences for the SMR.
- Also the industry-driven NEI 'White Paper' on SMR EPZ (2013) observes NUREG-0396 and SECY-97-020 indicate that the margins of safety provided by the 10-mile EPZ for existing plants were not based on quantification of accidents, but rather 'were qualitatively found adequate as a matter of judgment'.

Sandia National Laboratories (2013) published an extensive 186-page report on Evaluation of the Applicability of Existing Nuclear Power Plant Regulatory Requirements in the U.S. to Advanced Small Modular Reactors in which the applicability of current regulations to advanced SMRs is examined:

'*In a NUREG-0654 footnote, it is stated that the radii (plume 10 and ingestion 50 miles) are applicable to light water nuclear power plants, rated at 250 MWt or greater. Small water cooled power reactors (less than 250 MWt) and the Fort St. Vrain gas cooled reactor may use a plume exposure emergency planning zone of about 5 miles in radius and an ingestion pathway emergency planning zone of about 30 miles in radius.The alert and notification system will be scaled on a case-by-case basis. These are based on the lower potential hazard from these facilities, with lower radionuclide inventory and longer times to release significant amounts of activity for many accident scenarios. A similar argument may be able to be used for SMRs.*'

About the existing emergency planning requirements, the Sandia (2013) report contains:

Applicability of Existing Emergency Planning Requirements to Advanced SMRs

- Regulations for NPP Emergency Planning
- Sub-tier Guidance Documents for NPP Emergency Planning
    - NUREG-0800: Standard Review Plan
    - RG 1.101: Emergency Response Planning and Preparedness for Nuclear Power Reactors
    - EPA-400-R-92-001: Manual of Protective Action Guides and Protective Actions for Nuclear Incidents
    - NUREGs
    - Other Guidance Documents
- Summary of Emergency Planning Requirements Applicability to Advanced SMRs

4.9 Zones & distances recommended by the IAEA

IAEA recommendations on emergency planning & zones are best found in GSR Part 7 (2015) & EPR-NPP_PPA (2013). Ramon de la Vega (Emergency Preparedness Coordinator of the IAEA Incident and Emergency Centre IEC, 2018) crystallized the IAEA roles and responsibilities in the following:

- Development Safety Standards and guidance and tools on Emergency Preparedness and Response (EPR)
- EPR capacity building
- Provision of Peer Review Services: Emergency Preparedness Review (EPREV)
- IEC is the global focal point for emergency preparedness and response for nuclear and radiological safety or security related emergencies, threats or events and world's centre for coordination of international emergency preparedness and response assistance.

The IAEA has defined for nuclear facilities 5 threat categories, of which the first (I) is the worst, with potential of causing severe deterministic health effects offsite. In category II, such health effects are not expected offsite, and in category III, the need for protective actions is restricted to on-site only. The estimated potential offsite consequences for SMR plants have a range from category I to category III, so that reduction of EPR arrangements, compared with large LWRs, could be possible. In any case developers and operators must prove the improved safety.

Some relevant IAEA Requirements and Guides on dispersion and dose assessment and protective radiological countermeasures include GSR Part 7, GSG-2, GS-G-2.1, and NS-G-3.2. Of these, clearly the newest and most important one is GSR (General Safety Requirements) Part 7, Preparedness and Response for a Nuclear or Radiological Emergency, from year 2015. The requirements are explained in more detail and justified in the IAEA 2013 report 'EPR-NPP Public Protective Actions' (Actions to protect the public in an emergency due to severe conditions at a light water reactor), which can be considered as practical guidance in IAEA EPR Series. Notably it includes also the dose to a foetus.

Some IAEA standards / publications relevant for SMR EPZ determination are listed below:

- TECDOC-953, Method for developing arrangements for response to a nuclear or radiological emergency, EPR-method 2003, October 2003
- GSR Part 7 (136 pages, 2015, supersedes GS-R-2 of 2002):
  Preparedness and Response for a Nuclear or Radiological Emergency
- GSG-2 (120 pages, 2011):
  Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency
- GSG-11: Arrangements for the termination of a nuclear or radiological emergency

- GS-G-2.1 (159 pages, 2007):
  Arrangements for Preparedness for a Nuclear or Radiological Emergency
- NS-G-3.2 (42 pages, 2002):
  Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for NPPs
- EPR-NPP-OILs, 160 pages, 2017 (Operational Intervention Levels)
- EPR-NPP-PPA, 159 pages, 2013 (Public Protective Actions)
- IAEA 'SMR Book', 2020 Edition
- IAEA SMR Regulators' Forum pilot project report (2018)

IAEA GSR Part 7 (Preparedness and Response for a Nuclear or Radiological Emergency) from 2015 (136 pages) supersedes GS-R-2 and contains requirements on the following aspects of EPR:

- Ensure adequate level of preparedness & response
- 26 numbered requirements
- Responsibilities, management, notification, protective actions, public information, emergency workers, medical response, wastes, logistics, exercises, restricting doses
- GSR Part 7 is technology neutral: applicable to all facilities and activities, fully including Next Generation Reactors (NGR).
- Hazard Assessment as defined in GSR Part 7 (includes Graded Approach) should be applied to define appropriate requirements (including EPZ) for NGR EPR.
- It is not appropriate to consider EPR as a design issue (i.e. as being mainly related/influenced by the design safety).

IAEA EPR-NPP PPA (Actions to protect the public in an emergency due to severe conditions at a light water reactor, 2013):

- Overview document for off-site decision makers in plain language
- Latest guidance on public protective actions for emergencies caused by SA at LWR
- Considerations for EPZ sizing for LWR
- Basic concepts (taken from GSR-Part 7) may be applied to all kinds of designs
- Suggested sizes for EPZ and EPD may differ for other kind of reactors, particularly for reactors with enhanced safety features like SMR (According to Vilar-Welter of IAEA, no commonly accepted, technically sound methodology exists to date to define size of EPZ/D to SMR.)
- Section 4: Offsite emergency zones and distances
- Section 5: Urgent protective actions and early protective actions and other response actions
- Appendix I: Basis for the suggested size and protective actions within the emergency zones and distances

The zones related to EPR (emergency preparedness & response) and longer-term actions can be explained as follows. Note that only in PAZ are severe deterministic health effects possible. In the farther zones, there may only be stochastic health effects:

- PAZ (precautionary action zone): Preparedness for precautionary urgent protective actions (before release or shortly after it begins) to reduce the risk of severe deterministic effects. Extends e.g. to 5 km from a typical large power reactor.
- UPZ (urgent protective action planning zone): Preparedness for urgent protective actions to be taken promptly to avert offsite doses. Extends to 25 km from a typical large NPP. Note: In Finland, the term 'Emergency planning zone' (EPZ), extending to 20 km from reactor, was usually used.
- LPZ (longer-term protective action zone), also called FRPZ (food restriction planning zone): Preparedness for protective actions to reduce the long-term dose (stochastic health effects from groundshine and ingestion of local food). The LPZ may extend to 300 km from a large NPP.
- EPD (extended planning distance): Monitor the situation to find areas in which response actions would be needed within the time period 1 d to a few weeks. E.g. 100 km.

- ICPD (ingestion and commodities planning distance): Reduce stochastic effects due to contaminated food, milk and drinking water, and commodities other than food. E.g. 300 km.

The EPR-NPP_PPA has in Appendix I (‘Basis for the suggested size and protective actions within the emergency zones and distances’) some sample zone size calculations of the PAZ and UPZ, starting from ground level release of 10 % of the volatile fission products in a 3000 MWth NPP. Variable release rate for the 10 h duration and the meteorological conditions (neutral stability D) used are given. Public behavior and various protective countermeasures are accounted for. Table 19 of EPR-NPP_PPA shows the dose limits and exposure pathways used. As the Appendix I clearly admits, the calculations shown there are a ‘first approximation’: very uncertain and based on very simple assumptions. They may (and actually should!) be modified for a specific power plant and local conditions.

A first and easy approximation of possibly suitable zone sizes for SMR can be made with IAEA GS-G-2.1 (Arrangements for preparedness for a nuclear or radiological emergency) or the EPR-NPP PPA, which recommend, as a ‘rule of thumb’, for e.g. 160 MWth (cf. Table 5 and Table 6) quite broad intervals. Note the increase of zone radii after the Fukushima accident of March 2011:

- PAZ (precautionary action zone) = 0.5 … 3 km, or 3…5 km
- UPZ (urgent protective action zone) = 5 … 30 km, or 15…30 km

Table 5. Suggested (in 2007) emergency zones and area sizes for nuclear facilities of different sizes. Source of Table: Table 8 of IAEA GS-G-2.1 (2007). Slightly different values were found in TECDOC-953 (Appendix 5, Table A5-II).

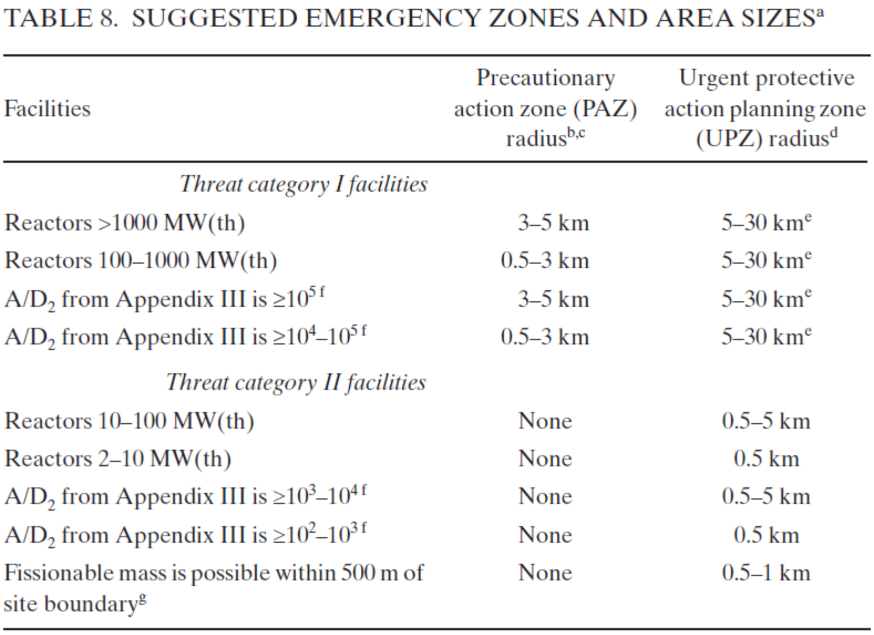
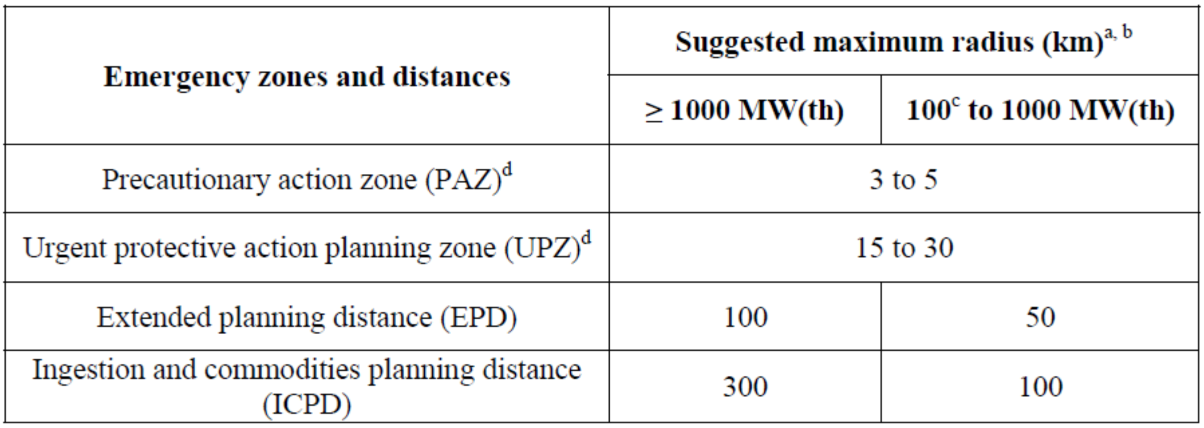


TABLE 8. SUGGESTED EMERGENCY ZONES AND AREA SIZES[a]

Facilities	Precautionary action zone (PAZ) radius[b,c]	Urgent protective action planning zone (UPZ) radius[d]
Threat category I facilities		
Reactors >1000 MW(th)	3–5 km	5–30 km[e]
Reactors 100–1000 MW(th)	0.5–3 km	5–30 km[e]
A/D_2 from Appendix III is $\geq 10^5$[f]	3–5 km	5–30 km[e]
A/D_2 from Appendix III is $\geq 10^4$–10^5[f]	0.5–3 km	5–30 km[e]
Threat category II facilities		
Reactors 10–100 MW(th)	None	0.5–5 km
Reactors 2–10 MW(th)	None	0.5 km
A/D_2 from Appendix III is $\geq 10^3$–10^4[f]	None	0.5–5 km
A/D_2 from Appendix III is $\geq 10^2$–10^3[f]	None	0.5 km
Fissionable mass is possible within 500 m of site boundary[g]	None	0.5–1 km

Table 6. Suggested (in 2013) emergency zones and area sizes for nuclear facilities of different sizes. Source of Table: Table 3 of IAEA EPR-NPP_PPA.



Emergency zones and distances	**Suggested maximum radius (km)[a, b]**	
	≥ 1000 MW(th)	**100[c] to 1000 MW(th)**
Precautionary action zone (PAZ)[d]	3 to 5	
Urgent protective action planning zone (UPZ)[d]	15 to 30	
Extended planning distance (EPD)	100	50
Ingestion and commodities planning distance (ICPD)	300	100

The Fukushima accident showed that for sites with large LWR, the sizes of emergency planning zones and distances should be larger than previously thought. After Fukushima, the EPD (extended planning distance) and ICPD (ingestion and commodities planning distance) were clearly defined at appr. 100 km and 300 km, respectively. In this work the emphasis is on the first two: PAZ and UPZ, which have biggest emergency significance.

5. International SMR-specific development of EPZ

5.1 EPZ sizes claimed by SMR plant providers

This Chapter contains:

- SMR plant providers' comments / justifications on their needed EPZ
- Attempted guidelines by organizations representing the industry, like NEI (Nuclear Energy Institute) and possibly also EPRI (Electric Power Research Institute)
- Design-specific EPZs mentioned in the literature
- As a special example, the licensing process of NuScale & TVA in the US

The IAEA has published unofficial SMR Handbooks, the latest being the 2020 Edition of 'Advances in Small Modular Reactor Technology Developments', which contain up-to-date available basic design data of more than 50 different SMR designs (water-cooled, HTGR, fast neutron, molten salt). The Handbook has the plant footprint (m2) for most designs, but not the designer-proposed EPZ figures. Table 7 below ('Emergency zones and distances mentioned by designers') is based on a presentation during an IAEA technical meeting in February 2017.

Table 7. Emergency zones and distances mentioned for some SMR designs. Source of data: IAEA technical meeting in February 2017. Note: The data are not official and may contain errors or be subject to change.

Zones and distances	**SMR designs**
No off-site EP plan	VK-300, GT-MHR, 4S
Simplified EP plan	CAREM-25, mPower, NuScale, CCR, HTR-PM, G4M
400 meters	PBMR
1000 m, no off-site evacuation	KLT-4S, VBER-300, ABV
1500 m	SMART
2000 m	IRIS
Not specified	IMR, GTHTR300, PASCAR

Ramana et al. (2013) provide a survey of SMR licensing in the US, Russia, China, India and South Korea. They observe that SMR designers have sought to get one or more typical licensing requirements for large reactors 'diluted', e.g. get regulatory authorities to eliminate the requirement for an EPZ or at least reduce its size. However, there are variations in the way different countries have treated the issue of EPZ size.

For the SMART design of South Korea, Ramana et al. (2013) include the following historical information on EPZ: 'As in other countries, the question of what size EPZ is appropriate to SMART was raised during the license reviewing, but it was decided that the EPZ of SMART must be the same as that of a commercial power reactor. However, SMART's designers would like its EPZ to have a radius of 1.5 km and the LPZ (low population zone) radius to be 2 km. According to KAERI, the area needed for two units of SMART is about 126,000 square meters (radius 200 m) including an EAB (Exclusionary Area Boundary).' They conclude that 'In South Korea the vendor of the SMART was apparently less concerned with the size of the EPZ'.

In the case of SMR EPZ in China, Ramana et al. (2013) conclude that 'SMR developers have tried to reduce the size of the EPZ, perhaps in the hope of eliminating it altogether in the future'. They have gathered the following information for two SMR designs:

- HTR-PM: Project proponents note that technically 'off-site emergency planning measures can be simplified remarkably'. However, for the first set of HTR-PMs, its designers have adopted a different strategy. While talking about 'the technical possibility that HTR-PM can eliminate offsite emergency planning', they have chosen to co-locate these reactors with large light water reactors (LWRs) so that the latter's EPZ requirements would dominate. The obvious result of this strategy is that there would be no need for the developers of the HTR-PM to enter into a debate over whether the EPZ for the HTR-PM should be reduced.
- ACP100: One of the aims is 'to eliminate the emergency evacuation zone of an ACP100 nuclear plant' so that in the 'near future, nuclear plants can be built right next to cities'. Project proponents do advertise the reactor as requiring a 'smaller emergency off-site area'.

In general, Ramana et al. (2013) describe the EPZ claims of SMR vendors: 'Even the lower distance figure of 5 km that is suggested by the IAEA, while smaller than the EPZ size typically chosen for current standard sized reactors, is not acceptable to most SMR developers and potential operators. Many SMR vendors have advertised much smaller EPZ radii, arguing that the improved safety of SMRs is sufficient reason to lower the size of an EPZ. For example, Babcock and Wilcox (J Ferrara 2012) lists a small EPZ radius, down to 1000 feet, as one of the ten game-changers offered by its mPower reactor. Likewise, NuScale's website announces that the design of the whole power plant will support a reduced Emergency Planning Zone based on small core size and use of the mechanistic source term methodology.'

Reduced EPZ may give rise to negative public opinions, like the UCS (2013): 'Generation mPower states that it can reduce the EPZ radius for its 360 MWe twopack down to a mere 1,000 feet (Generation mPower 2013), a distance that could be inside the power plant site boundary. It has not explained the basis for this reduction.'

Ramana & Mian (2014) describe the IRIS design by Westinghouse: 'Many SMR designers, on the other hand, would like to avoid evacuation planning altogether. For example, one risk-related goal of the now-abandoned IRIS reactor designed by Westinghouse was to reduce the Emergency Planning Zone (EPZ) to within the exclusion area by demonstrating that the off-site doses are consistent with the US Protective Action Guidelines (PAGs) for initiation of emergency response, so that the required protective actions would be limited to the exclusion area.'

In Japan, Toshiba offers the 4S ('Super-safe, Small and Simple') sodium-cooled pool-type fast reactor with 30 MWth power. They have studied the possibility of EPZ radius of 200 m by calculating total effective dose equivalents (TEDE) vs. distance.

The Idaho National Laboratory (INL, 2014) published a review 'Opportunities in SMR emergency planning', where they picked four particularly relevant documents:

- SECY-11-0152, Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors (NRC, 2011)
- The industry-driven NEI (Nuclear Energy Institute) EP/EPZ 'White paper': Proposed Methodology and Criteria for Establishing the Technical Basis for Small Modular Reactor Emergency Planning Zone (NEI, 2013)
- Sandia National Laboratories: Evaluation of the Applicability of Existing Nuclear Power Plant Regulatory Requirements in the U.S. to Advanced Small Modular Reactors (Sandia, 2013)
- INL HTGR EPZ White Paper (INL/MIS-10-19799); INL, 2010

INL (2014) describes the NEI (2013) White paper on SMR EPZ: 'Using an expectation of enhanced inherent design safety, emergency preparedness regulatory framework and dose savings criteria, and the body of risk information which will be developed for each plant in a final safety analysis report, the white paper discussed a generic methodology and criteria to establish a design- and site-specific technical bases for determining the appropriate EPZ size for a SMR.'

In NEI (2013), the nuclear industry is 'proposing to apply the NUREG-0396 sizing rationale to SMR EPZ size determination (see Fig. 5), and at the same time to apply the significant body of severe accident information that has been developed in the over three decades since NUREG-0396 was published (i.e. 1978), and to apply the design specific and plant-specific PRA information that will be prepared to support SMR licensing'.

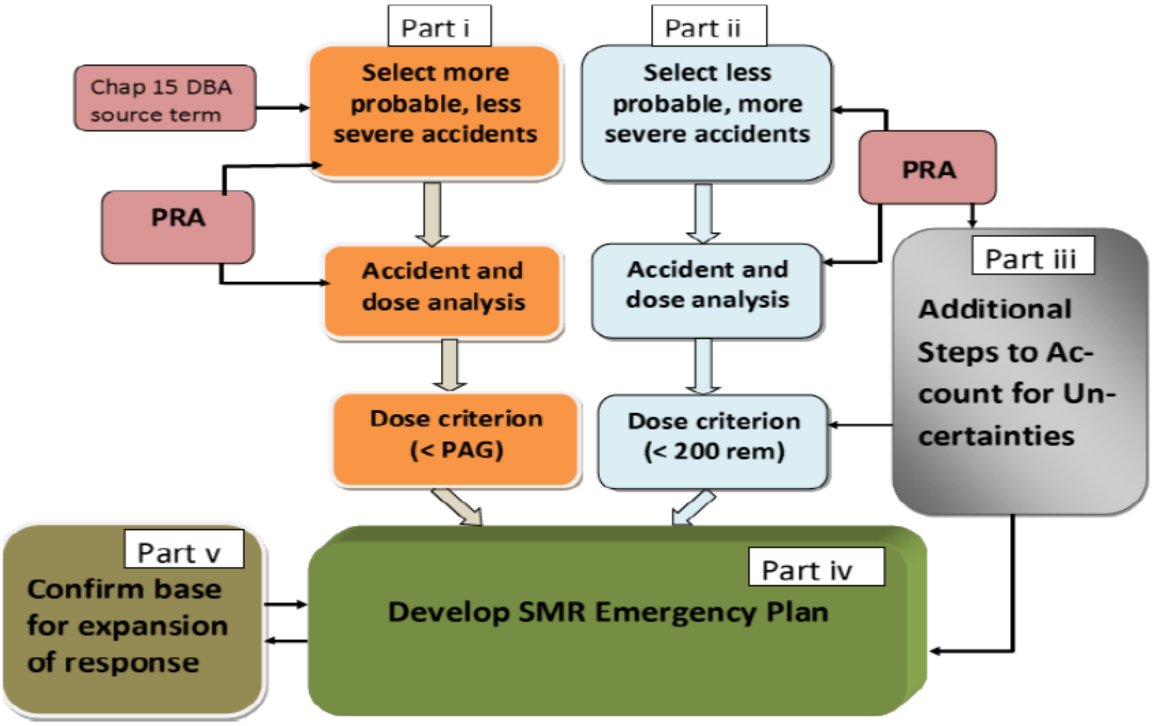


Figure 5. The 'Integrated SMR EP Approach', suggested by the Nuclear Energy Institute (NEI), including the technical criteria for EPZ sizing. Source of Figure: NEI (2013).

The NRC has published its feedback questions on the NEI-proposed SMR EPZ methodology (ML14142A425.pdf). As these regulator-side questions are particularly interesting for the purposes of this report, they are listed here below for easy reference. Some of the questions are also expanded / explained here:

1. *What are the key design features?*
2. *Will applicants arrive at sufficient conclusions (using this)?*
3. *Fukushima lessons learnt? Multi-unit accident?*
4. *Melcor, Maap5, Soarca extrapolated for SMR?*
5. *Risk-informed judgement / How will insights be used?*
6. *Ways of using PRA for accident scenario selection? Multi-module core damage source term?*

7. The paper states that the MELCOR Accident Consequence Code System (MACCS) code is an appropriate tool to calculate consequences for the analyses. The paper also states that insights from the NRC's use of MACCS for the SOARCA will be used to inform use of the code for this purpose.

a. Considering that MACCS has not previously been used for SMR analyses to support EPZ sizing, will NEI provide more information or guidance on the use of the MACCS code for this purpose? Topics that *could be different for this proposed use of MACCS are the determination of the basis for code input and model assumptions (including appropriate nodalization of the near-field), sources of information for input, use of conservatism, addressing uncertainty, and input and assumptions for the local area and population.*

b. Which information from SOARCA is proposed to be used, and how will it be used?

c. Will a test case or pilot case be provided for demonstrating how MACCS and SOARCA information would be applied in a realistic situation?

8. Evacuation modeled?
9. Will accident scenarios be grouped?
10. SOARCA >> cut-off frequency 1e-8 / plant-year?
11. Fuel handling & spent fuel pool accidents?
12. Reactor-year vs. plant-year (coincident core damage events)
13. Source terms not additive for multi-module SMRs?
14. Basis for accident sequence selection: 5 docs adapted for SMRs?
15. Methodology for determining the probability of dose exceedance?

16. What is the proposed probability basis for Criterion c (probability of dose exceedance)?
Is it probability over
-weather trials;
-over scenarios;
-over accident classification (frequent, infrequent, severe);
over type (internal, external, low power and shutdown, internal flood, internal fire, other);
-over release categories;
-or something else?

17. Conditional or absolute probability of exceeding a whole body acute dose of 2 Sv?
18. Technical adequacy of the base PRA? Relevant initiating events?
19. Accepted uncertainty in CDF and LERF?
20. Smaller core changes acceptance for CDF and LERF?
21. Application of regional assets in FSAR?
22. Mitigation strategies in level 2 PRA?
23. Risk insights & DiD considerations >> plume exposure SMR EPZ size?
24. Plant simulator to address the uncertainties (control room, staffing, procedures, EP) ?
25. Low-freq - high consequence events: several options?
26. What is 'enhanced plant capabilities'?
27. Qualitative and quantitative approaches to decrease current 10 mile plume EPZ

Talabi & Fischbeck (2016), of Pittsburgh Technical LLC & Carnegie Mellon University, presented their support to the nuclear industry in ITMSR-4: 'The study supports the nuclear industry's approach to secure US Nuclear Regulatory Commission (USNRC) approval of a scalable Emergency Planning Zone (EPZ) specific to light-water cooled Small Modular Reactors (SMR). The study describes the approach to achieve regulatory approval, which requires demonstrating a reduced SMR risk profile, by establishing SMR-specific Probabilistic Risk Assessments (PRA) through the description of methodological and technological advancements that reduce likelihood of accidents and potential dose.'

The EPRI (Electric Power Research Institute, 2016), in its Advanced Nuclear Technology (ANT) program, studied EPZ size evaluation for SMRs. The results are available at no cost to funding members only:

- Gap assessment of SMR EP
- SMR EP roadmap
- Assessment of atmospheric dispersion models
- Proposed approach to dose calculation methodology
- Quantifying evacuation time reduction

NuScale, an option for TVA's Clinch River site:

One of the most advanced processes of determining right-sized EPZ for SMR can be found in the USA, by NuScale Power, whose SMR is one option for the TVA (Tennessee Valley Authority) Clinch River site, near the ORNL (Oak Ridge National Laboratory). The NuScale SMR plant can be characterized as follows: Reactor type IPWR (Integral PWR), electric output 50 MW (gross), primary circulation natural, 1-12 reactor modules per plant, and predicted core damage frequency from internal events 1e-8 per reactor-year, among other data.

NuScale uses natural circulation in various applications of generating coolant flow, but there are also many other features using passive safety (inherent features or safety systems). The probability and magnitude of atmospheric radioactive releases is basically dependent on their abilities. The NuScale SMR was more thoroughly described in VTT-R-05548-16 [Hillberg et al. 2016], in which the reader may find more information.

NuScale (2018) tells on their website about the efforts for 'rightsizing the EPZ': NuScale Power worked with the Nuclear Energy Institute (NEI) to develop methodologies to support smaller EPZs for SMRs for discussion with the NRC. In December 2015, NuScale submitted a topical report (NuScale_2015_licensing_topical_report_ML15328A088.pdf) to the NRC detailing its proposed methodology for NuScale Plant licensees to determine the appropriate EPZ.

Quite interesting results could be available in the LTR report given by NuScale Power to USNRC: NuScale Licensing Topical Report (LTR) on Design-Specific Emergency Planning Zone Sizing Methodology, available from NRC web pages as ML15328A088.pdf. However, that is the nonproprietary version and the most interesting results have been left blank, so a thorough comparison is not possible. The objective of the LTR report was to provide for NRC review the technical basis for the plume exposure EPZ sizing methodology for the NuScale design. NuScale Power requests that the NRC would provide a SER (safety evaluation report) on the design-specific EPZ sizing methodology, concluding that the proposed methodology is an acceptable approach for justifying the EPZ size for the NuScale design. In short, the LTR contains:

- Accidents to be evaluated
- DBA & 2 classes of severe accidents
- Multi-module risks; risks outside PRA
- Source-term & dose evaluations:
- MACCS2 code was used
- Mean TEDE (total effective dose equivalent) vs. distance from reactor building
- Site meteorological conditions leading to highest doses

TVA (Tennessee Valley Authority, an electric etc. utility / agency) submitted to the NRC an ESP (Early Site Permit) application for two or more SMR modules in May 2016. The application is for the Clinch River Nuclear (CRN) Site in Oak Ridge, Tennessee, and is based on a plant parameter envelope encompassing the light-water small modular reactors (SMR) currently under development in the United States by BWX Technologies, Holtec, NuScale Power, and Westinghouse.

In July 2018, the NRC published a staff audit report (ML18177A107.pdf) 'Summary report for the second regulatory audit of Clinch River nuclear site Early Site Permit application - Part 6 Exemptions and departures, emergency planning exemptions'. As the summary of NRC observations is particularly interesting for the purposes of this report, it is included here in full text:

Based on the NRC staff's audit of the applicant's documentation of representative PEP EPZ size determination analysis, determination of related plant parameter accident release source terms, and related calculations and analyses, the staff observed the following:

1. Through review of the proprietary NuScale calculation package in the eRR, the staff was able to understand more fully the inputs, assumptions and methodology used in the representative analysis of the *consequences of accidents at the site boundary for the NuScale design. The staff noted that the NuScale analysis used design information that is consistent with the current state of the NuScale design certification review, and conformed to the methodology presented in CRN SSAR 13.3 for evaluation of the EPZ size. The NuScale calculation also included sensitivity analyses that varied modeling assumptions and inputs related to atmospheric transport and dispersion and CRN site specific information. The analysis showed that for the current information for the NuScale design, which is evaluated as a representative SMR within the ESPA PPE, the dose consequences of design basis and more probable severe accidents would be less than the TVA SSAR Section 13.3 EPZ size dose criteria at the CRN site boundary. Therefore, based on the staff's improved understanding of the representative analysis that supported TVA's response to RAI Letter No. 10, eRAI-9206, Question 1, this audit item is closed.*

2. NuScale calculation ER-P030-5335 was updated in Revision 2 to include updated NuScale design information and add Appendix G to describe the development of a nondesign-specific plant parameter atmospheric release source term related to assessment of the EPZ size exemptions. This plant parameter source term was presented in Table 5, "EPZ PPE Source Term," in Enclosure 1 of TVA's response to RAI Letter No. 10, eRAI-9206, Question 1, dated March 30, 2018. The staff's review of Appendix G focused on the explanation of the development of non-design-specific plant parameter atmospheric release source term.

The staff observed that the development of the non-design-specific plant parameter source term included accident isotopic release information (three source terms) from different sources; (1) the PPE bounding SSAR Chapter 15 DBA atmospheric release source term, and (2) DBA and severe accident atmospheric releases for the NuScale design used to evaluate EPZ size taken from the main body of NuScale calculation ER-P030-5335 (i.e., the representative analysis). The staff observed that the method used to develop the non-design-specific plant parameter atmospheric release source term was consistent with the description in Enclosure 1 of TVA's response to RAI Letter No. 10, eRAI-9206, Question 1, dated March 30, 2018. The analysis determined a composite 96-hr total activity per nuclide release source term based on the maximum per nuclide release from three source terms described above, with an additional margin of 25 percent (TVA composite + 25% source term). Then, the TVA composite + 25 percent was used as the basis for input to a series of input cases to the MELCOR Accident Consequence Code System (MACCS) radiological consequence code to determine a maximized atmospheric release source term that would meet the EPZ size methodology dose criteria at the site boundary. Through review of the MACCS analysis cases, the staff noted that the nuclide activity release inputs were adjusted to ensure that the total nuclide release activities in MACCS analysis Case D that was chosen as the "EPZ PPE Source Term" was bounding for the TVA composite + 25 percent atmospheric release source term. The staff noted that the MACCS analysis cases added additional release margin, while remaining below the EPZ size dose criteria at the site boundary, by approximately 10 percent for the selected Case D.

The staff observed that the MACCS Case D released activity from containment reported in ER-P030-5335, Revision 2, Appendix G, is the same as was provided in Table 5 in Enclosure 1 of TVA's response to RAI Letter No. 10, eRAI-9206, Question 1, dated March 30, 2018. The staff notes that the method used to develop the "EPZ PPE Source Term" considered accident release information from a range of accidents (both DBAs and severe accidents) and from a range of SMR designs considered in the CRN ESPA, including the smallest core power (NuScale) and the largest core power (Westinghouse SMR, as the ESPA PPE accident source term). The staff also notes that the method used to develop the "EPZ PPE Source Term" included conservatism both in the amount of activity assumed to be released to the atmosphere and in the margin to the EPZ size dose criteria at the site boundary. Therefore, based on the staff's improved understanding of the development of the non-design-specific plant parameter atmospheric release source term related to the EPZ exemption requests, this audit item is closed.

In August 2018, the NRC gave a favorable ruling, stating that the TVA's safety methodology can be used for re-sizing the EPZ around an SMR plant. According to a piece of news in Nuclear Energy Insider (19 September, 2018):

‘Last month, Nuclear Regulatory Commission (NRC) staff concluded that safety methodology submitted by Tennessee Valley Authority (TVA) could be used to determine the size of emergency planning zones (EPZs) required for Small Modular Reactors (SMRs).

The staff found TVA’s proposed dose-based, consequence-oriented methodology to be a “reasonable technical basis” for determining EPZ size, consistent with the basis used to determine EPZ size for large light water reactors.

The agency also granted TVA its exemption from a 10-mile EPZ for future combined construction and operating license applications for which the radioactive source term is bounded by the conditions established by the NRC. A July 2018 staff audit report found that an SMR plant at the Clinch River site based on the NuScale SMR design would meet the conditions for a site boundary-sized EPZ.

Application of the methodology would end the blanket mandatory 10-mile minimum EPZ requirement, put in place for larger conventional LWRs. In its ESP, TVA has proposed a 2-mile EPZ or site boundary EPZ, depending on design specifications.’

5.2 US NRC attitude to scalable EPZ size

In the US, the most central legislation for licensing is 10 CFR Part 50 (Domestic Licensing of Production and Utilization Facilities): Title 10 of the Code of Federal Regulations, one of fifty titles comprising the United States Code of Federal Regulations (CFR), containing the principal set of rules and regulations issued by federal agencies regarding nuclear energy. A two-step process (construction permit & operating license) has been used.

NRC work on SMR smaller EPZ (already partly described above in Ch. 5.1) has been centered around the question: Which is cheaper and more efficient way of working, providing case-by-case decisions (‘option 1’ below), or making clear rules that can obeyed by licensees before they submit an application (‘option 2’). The result and current work (NRC 2017 draft regulatory basis for SMR EP) seems to be the latter option, or rule-making:

- Option 1 (exemptions & guidance)
- Option 2 (conduct rulemaking): stability, predictability, clarity; recognize advancements; credit small size

There have been claims that the DOE is ‘pressing’ the NRC to make licensing regulations easier for SMR. The purpose would be to get US domestic SMR designs licensed and built quickly enough to gain markets from SMR vendors of other countries. For example, UCS (2013): ‘In its March 2013 second solicitation for its SMR licensing support program, the DOE states that it is looking for designs that present a credible case to the NRC to reduce emergency preparedness zone requirements.’

According to www.nrc.gov, the SECY papers at NRC are Commission Papers, or written issues papers the NRC staff submits to the Commission to inform them about policy, rulemaking, and adjudicatory matters. In the 2017 draft regulatory basis (ML16309A332) ‘Rulemaking for emergency preparedness for small modular reactors and other new technologies’, several relevant SECY papers are listed in historical order:

- SECY-93-092: advanced reactor design issues & current regulatory requirements
- SECY-97-020: emergency planning for evolutionary & advanced reactors
- SRM-SECY-04-0236: common emergency operating facility at corporate headquarters
- SECY-06-0200: review of emergency preparedness regulations and guidance
- SECY-10-0034: policy, licensing and technical issues for SMR designs
- SECY-11-0152: EPR framework for SMR. The NRC has licensed several small reactors with a reduced EPZ of 5 miles.

- SECY-14-0038: performance-based framework for NPP EP oversight
- SECY-15-0077: options for EP for SMR & ONT
- SECY-16-0069: rulemaking plan on EP for SMR & ONT

Under the NRC website (https://www.nrc.gov/reactors/new-reactors/regs-guides-comm/ep-smr-other.html), 'Emergency preparedness rulemaking with regard to small modular reactors and other new technologies', the historical evolution of SMR regulation is explained in more detail:

- 10/1975: NUREG-75/014, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants (WASH-1400)
- 12/1978: "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," NUREG-0396/EPA 520/1-78-016 (NUREG-0396). In addition to identifying the recommended EPZ sizes for emergency planning, NUREG-0396 stated the following: The range of possible selections for a planning basis is very large, starting with a zero point of requiring no planning at all because significant offsite radiological accident consequences are unlikely to occur, to planning for the worst physically possible accident regardless of its extremely low likelihood. According to NEI (2013), the NRC supports maintaining the NUREG-0396 approach, as indicated in SECY 97-020 and SECY-11-0152, which state that the current rationale for the size of the EPZ, i.e., potential consequences from a 'spectrum of accidents' tempered by probability considerations, should be maintained.
- In the staff requirements memorandum (SRM) for SECY-93-092, the Commission directed that 'the staff should submit to the Commission recommendations for proposed technical criteria and methods to use to justify simplification of existing emergency planning requirements'.
- SECY-97-020: 'Because industry has not petitioned for changes to EP requirements for evolutionary and passive advanced LWRs, the staff did not dedicate the resources to fully evaluate these issues.
- 3/2010, the staff noted in SECY-10-0034 that EP is a key technical issue for licensing both SMRs and non-LWRs because of its role in the protection of the public, as well as its relationship to the key technical issue of SMR accident source term. The staff recognizes that EP requirements, particularly those concerning potentially reduced EPZ sizes, are important in decreasing regulatory uncertainty for SMR design certification and combined license applications. Also, industry has indicated that the EPZ issue will be a key factor in the business case for SMR feasibility and development.
- SECY-11-0089: 'The NRC is developing a fullscope, level 3 PRA which will reflect current state-of-practice methods, tools, and data, and will incorporate technical advances since the last NRC-sponsored Level 3 PRAs which were completed over 20 years ago.'
- SECY-11-0152: (Development of an EPR framework for SMRs, ML112570439.pdf); This information paper discussed 'the staff's intent to develop a technology-inclusive, dose-based, consequence-oriented EP framework for SMR sites that takes into account the various designs, modularity and collocation, as well as the size of the EPZ'. NRC has already licensed various facilities with a reduced EPZ size, including several small reactors, fuel facilities, material facilities, and independent spent fuel storage installations. These facilities relied on a dose/distance approach to establish the boundary of their planning areas based on U.S. Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs) (EPA 1992).
- SOARCA (State-of-the-Art Reactor Consequence Analyses), NUREG-1935, final version 2012: SOARCA's main findings fall into three basic areas: how a reactor accident progresses; how existing systems and emergency measures can affect an accident's outcome; and how an accident would affect the public's health. The project's preliminary findings include: Existing resources and procedures can stop an accident, slow it down or reduce its impact before it can affect public health. Even if accidents proceed uncontrolled, they take much longer to happen and release much less radioactive material than earlier analyses suggested. The analyzed accidents would cause essentially zero immediate deaths and only a very, very small increase in the risk of long-term cancer deaths.

- 12/2013: The Nuclear Energy Institute (NEI) submitted a white paper to the NRC in December 2013 outlining a high level approach to determining SMR EPZ size.
- 8/2014: COL/ESP-ISG-027, Specific Environmental Guidance for Light Water Small Modular Reactor Reviews: Provides clarification of guidance related to review of environmental applications, like issues that arose during pre-application discussions with SMR vendors and applicants. Less detailed review for lower levels of impact.
- SECY-15-0077, "Options for Emergency Preparedness for Small Modular Reactors and Other New Technologies" (ADAMS Accession No. ML15037A176): Proceed with rulemaking for EP for SMRs and other new technologies.
- SECY-16-0069, "Rulemaking Plan on Emergency Preparedness for Small Modular Reactors and Other New Technologies": schedule for a rulemaking for emergency preparedness for SMRs
- 5/2016: Tennessee Valley Authority submitted an application for an early site permit (ESP) for two or more SMR modules (up to 800 MWe, 2420 MWt) at the Clinch River Nuclear site on May 12, 2016.
- 9/2017, ML17206A265.pdf (NRC-2015-0225): Final regulatory Basis - Rulemaking for EP for SMR and ONT, including e.g. Size of the EPZ and other offsite EP requirements; Source term, dose calculations, and siting; Collocation of facilities; Multi-module facilities; Performance-based approach to EP.

An interesting account, in chronological order, of NRC regulatory history of EPZ sizing can also be found in the ADAMS document ML18177A386.pdf (‘Emergency planning zone sizing for small modular reactors - Regulatory history and policy considerations’) written by Bruce Musico of NRC.

In the Federal Register 12 May 2020, the NRC proposed a new rule and guidance for SMR emergency planning and requested comments on it. NRC regulations would be amended to include new alternative emergency preparedness (EP) requirements for small modular reactors (SMRs). The technical basis was supported by 2018 research studies: ‘Required analyses for informing EPZ size determinations’ (ML18114A176) and ‘Generalized dose assessment methodology for informing EPZ size determinations’ (ML18064A317). Some highlights from the draft regulatory guide (DG-1350) include:

- Similar approach to the dose/distance rationale used historically by the NRC.
- Even the existing regulations allow to choose between 1) the 10 miles & 50 miles zones (PEP & IEP) or 2) follow a case-by-case EPZ size determination process under 10 CFR part 50.
- The range of potential SMR source terms and designs calls for an alternative scalable methodology for determining EPZ size on a case-specific basis.
- Generic methodology without design- or site-specific information (source term, fission products, projected offsite dose).
- NUREG-0396 (NRC 1978) established EPZs for large LWRs with the objective to provide dose savings for a spectrum of accidents.
- Applicants must establish their EPZ within which public dose is projected to exceed 10 mSv TEDE over 96 hours from the release resulting from a spectrum of credible accidents.
- In certain cases of SMR PEP EPZ extending beyond site boundary, the exact EPZ configuration should be determined with local emergency response needs and capabilities (affected by demography, topography, land characteristics, access routes, and jurisdictional boundaries).

In Appendix A of the NRC guide DG-1350, the NRC provides general guidance with the expectation that the industrial applicant will implement detailed design-specific calculations for NRC review and approval (but may also use some alternative well justified approach):

1. Source term (more detailed determination described in DG-1350 Appendix B)
2. Meteorological data development
3. Atmospheric transport model
4. Exposure model
5. Dose estimation
6. Probabilistic dose aggregation

To conclude from the documents listed above, the NRC will be changing EPR requirements for next generation reactors, using so-called graded approach. The NRC attitude to this is basically positive: The right-sized, scalable EPZ may be appropriate for SMR. The intent is to develop for light water SMRs an emergency preparedness framework which accounts for reactor design variations, modularity, co-location at same site, and EPZ size, and will be (cf. SECY-11-0152):

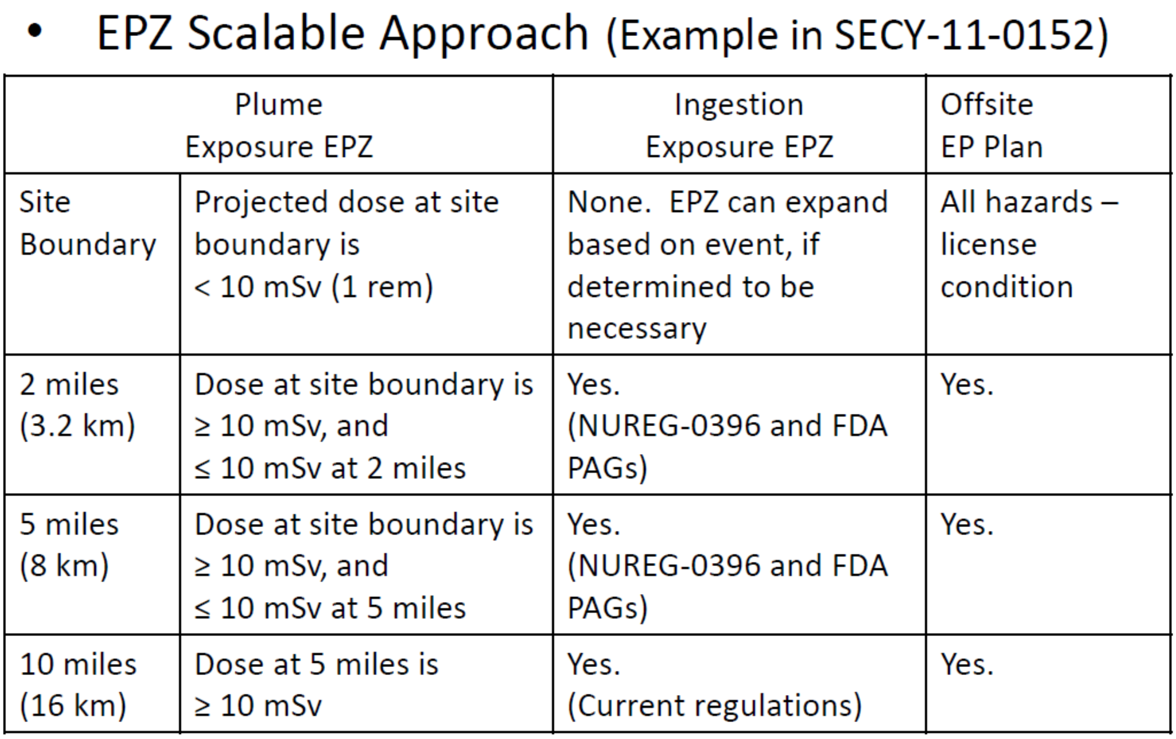
- Technology-neutral: The technology used for the reactor does not, by itself, affect EPR requirements, as long as the resulting safety level (frequencies and consequences) are the same.
- Dose-based: The measure of safety (and success of EPR) should be the doses received by individuals and population, together with their possible frequencies. (Note: This kind of information is traditionally represented by a CCDF, or complementary cumulative density function).
- Consequence-oriented: Note that there are also other consequences than radiological ones, if e.g. countermeasures (like evacuation) cause a lot of trouble, increased accidents, worsened medical care for those in need, etc.

Overall, the subjective and qualitative discussions about SMR safety should gradually shift towards more objective and quantitative work.

Regarding the costs caused by EPR from the utility point of view, using USA example, there are fees to NRC and FEMA (Federal Emergency Management Agency), capital costs and operating/maintenance costs (e.g. evacuation infrastructure, periodic training and equipment supply), which are affected by the size of the EPZ. From the NRC point of view, there are costs of rule-making and decision-making. It may be cheaper to prepare clear regulations than to decide about various exemptions for SMR plants in the future. With any strategy, societal aspects (perception of risk etc.) should also be taken into account.

Some information on current NRC work on SMR guidance is shown in Table 8 below (possible EPZ scalable approach, conditions for different EPZ radii).

Table 8. Possible EPZ scalable approach by NRC. Source: A.O. Costa, Presentation on NRC perspective of NGR (Next Gen) & EPR, IAEA 2017



- EPZ Scalable Approach (Example in SECY-11-0152)

Plume Exposure EPZ		Ingestion Exposure EPZ	Offsite EP Plan
Site Boundary	Projected dose at site boundary is < 10 mSv (1 rem)	None. EPZ can expand based on event, if determined to be necessary	All hazards – license condition
2 miles (3.2 km)	Dose at site boundary is ≥ 10 mSv, and ≤ 10 mSv at 2 miles	Yes. (NUREG-0396 and FDA PAGs)	Yes.
5 miles (8 km)	Dose at site boundary is ≥ 10 mSv, and ≤ 10 mSv at 5 miles	Yes. (NUREG-0396 and FDA PAGs)	Yes.
10 miles (16 km)	Dose at 5 miles is ≥ 10 mSv	Yes. (Current regulations)	Yes.

In addition to the NRC guide DG-1350 (May 2020) and other general guidelines, there are specific developments for non-LWR reactors, like molten salt reactors (MSR). In 2018 the Nuclear Energy Institute (NEI) published ‘Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development’ (ML18271A172). In April 2020 there was stakeholder discussion on the form and content of an application for non-LWR reactors (see ML20112F478). In March 2020, the industry-led Licensing Modernization Project (LMP), coordinated by NEI and cost-shared by DoE, published their final report. In June 2020, the NRC published their new RG 1.233 (Methodology for applications for non-LWRs). In March 2021, NRC published a compilation report (ML21085A484.pdf) about dose-assessment codes suitable for non-LWRs (but most codes equally well for other reactor types), for example SNAP/RADTRAD, GALE, RADTRAN, PAVAN, RASCAL or GENII.

5.3 Licensing and EPZ by CNSC (Canada)

Canada (cf. Chapter 10.6) is a good example of SMR-friendly licensing process and also provides many opportunities for SMR use in remote areas, mining sector etc. The size of the EPZ is flexible, thus suitable also for SMR graded approach, and use of full-scope PSA (levels 1-2-3) is encouraged (see pages 13-26 ‘Canada’ in the report of the IAEA Regulators’ Forum EPZ WG). A very brief description is given in the following (Figure 12 and text):

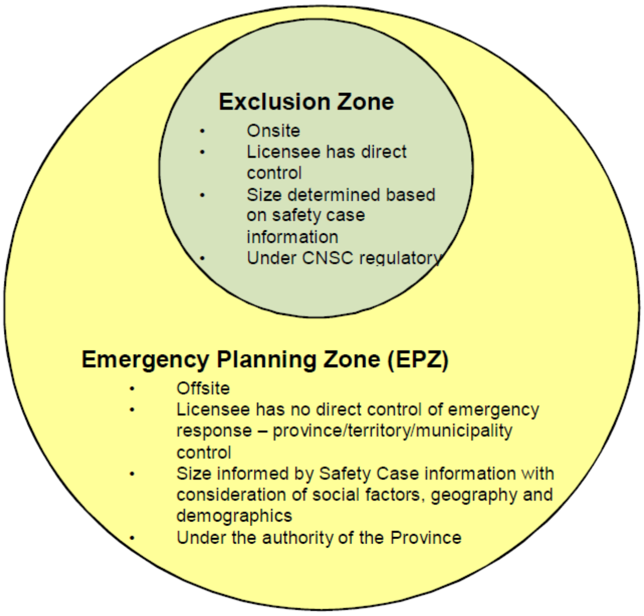


Figure 12. Relation of exclusion zone and EPZ in Canada. Source of Figure: IAEA SMR Regulators’ Forum EPZ WG report.

Figure 12 shows that in Canada, the exclusion zone (in licensee control) is under CNSC (the Canadian regulator; Canadian Nuclear Safety Commission) regulation and its size is based on the safety case, whereas the EPZ is under the province, with flexible size informed by the safety case (Example: Ontario updated Provincial Nuclear Emergency Response Plan, PNERP, 2017). The PNERP contains, according to CNL, the following descriptions and definitions:

- Operational Intervention Levels (OILs) and procedures for consequence mitigation
- Plant / site-specific Probabilistic Safety Assessment (PSA) based accident dose consequence evaluation
- 4 subzones of the EPZ around multi-unit CANDU plants:
    - Automatic Action Zone AAZ (3 km)

    - Detailed Planning Zone DPZ (10 km)
    - Contingency Planning Zone CPZ (20 km)
    - Ingestion Control Planning Zone (50 km)

Thus the EPZ in Canada consists of four subzones with their specific objectives, not explained in further detail in this report here. The EPZ is clearly recognized as part of DiD (see Figure 2 of the EPZ WG report). The role of the EPZ is considered in many phases: general requirements, license to prepare site, license to construct, operate and decommission.

Emergency planning in Canada is also supported by standards of the CSA (Canadian Standards Association), like the following:

- CSA-N288.2: Guidelines For Calculating The Radiological Consequences To The Public Of A Release Of Airborne Radioactive Material For Nuclear Reactor Accidents
- CSA-N290.15: Requirements for the Safe Operating Envelope of NPPs
- N290.16-16: Requirements for Beyond Design Basis Accidents (BDBA)
- CSA-N1600: General Requirements for Nuclear Emergency Management Programs

The EPZ WG describes the processes of determining EPZ and exclusion zone sizes in Canada:

- Process for determining EPZ extent in Canada:
    - Roles & responsibilities of provinces, Health Canada, CNSC, applicants
    - Physical design of reactor facility, PIEs, PSA 1-2-3
    - DSA (deterministic safety analysis), limiting credible accident
    - Source term and releases, meteorology
    - Dose assessment, dose criteria, external factors (town limits etc.)
    - EPZ determination process map
- Process for determining exclusion zone in Canada
    - PIEs, PSA, DSA, list of accidents
    - Limiting credible accident
    - EOPs (emergency operating procedures)
    - SAMG (severe accident management guidelines)
    - Source term & releases, meteorology, dose assessment, limits

The CNSC published in July 2018 their draft of REGDOC-1.1.5, ‘License Application Guide: Small Modular Reactor Facilities’.

According to the CNSC, ‘discussion papers’ play an important role in the selection and development of the regulatory framework and regulatory program. They are used to request early public feedback on CNSC policies or approaches. In 2016, the CNSC published on SMR regulatory approach a discussion paper, to which e.g. Ontario Power gave their comments, and then the CNSC published a ‘What we heard’ report. Finally in July 2018 a draft of ‘Licence application guide: SMR facilities’ (CNSC REGDOC-1.1.5) appeared:

- CNSC DIS-16-04: Small Modular Reactors: Regulatory Strategy, Aproaches and Challenges (CNSC_2016_DIS_16_04_SMR_Regulatory Strategy Approac.pdf, 44 pages):
    - The purpose is to engage interested stakeholders on the regulatory framework as it may apply to SMRs.
    - The paper presents a series of questions for the reader’s consideration:
    - Issues at a high level & short description of specific items to be addressed in future work
    - How the CNSC plans to address the issues using existing regulatory tools and processes
    - The implications of the innovative approaches being considered by SMR proponents that need to be examined to a greater degree, to confirm if additional supporting regulatory requirements or guidance are needed

- Ontario Power Generation's (OPG) comments on discussion paper DIS-16-04 (Ontario_Power_comments_on_SMR_regulatory_strategy_DIS-16-04-comment-received-OPG.pdf, 50 pages):
    - Multiple reactors on one site & licensing process for adding SMR modules
    - PSA, DiD
    - EPZ (Question 39): The OPG believe that 'the CNSC discussion paper covers the issue well, namely that there is already sufficient flexibility in the requirements for emergency planning zones so that no further regulatory guidance is needed.'

- CNSC_2016_What We Heard Report – DIS-16-04 - Canadian Nuclear Safety Comm.pdf (11 pages)

- CNSC, Licence application guide: SMR facilities, REGDOC-1.1.5, July 2018 (CNSC_REGDOC-1-1-5-licence-small-modular-reactor-facilities-draft-eng.pdf, 30 p.) contains, for Applicant's considerations by safety and control area, among other points, the following:
    - 2.2.4 Safety analysis
    - 2.2.7 Radiation protection
    - 2.2.9 Environmental protection
    - 2.2.10 Emergency management

To facilitate licensing of new NPPs, CNSC of Canada provides the possibility of VDR, or pre-licensing Vendor Design Review. The main idea is to provide feedback on the acceptability of design with respect to Canadian requirements to the plant supplier early in the design process, before submitting a license application to CNSC. The outcome of the VDR does not bind CNSC nor does it certify the design. The plant design to be reviewed could be traditional or advanced, small modular or large. On-going or completed VDRs as of 2021 include: IMSR, MMR, SEALER, ARC-100, Moltex, SMR-160, NuScale, U-Battery, BWRX-300 and Xe-100. NuScale made the first of four VDR submissions in January 2020, and the last of them was expected in November 2021.

In Canada, an 'SMR Roadmap' with 50 recommendations was published in 2018. Responding to it, a national SMR Action Plan appeared in 2020. Then in 2021, many pieces of news indicated the developing of an SMR ecosystem. The government will invest 56 million CAD to SMR research in New Brunswick. Canadian oil sands producers want to achieve net-zero greenhouse gas emissions, partly resorting to SMRs. Ontario Power Generation (OPG) will work with GE Hitachi to deploy a BWRX-300 at the Darlington site, to be completed in 2028, possibly as the first commercial SMR in Canada.

To conclude about the SMR opportunities in Canada, particularly the comment by Ontario Power about the 'already sufficient flexibility' in SMR EPZ requirements seems quite compelling, and possibly the licensing process in Canada could have some useful points for Finland as well.

CNL / Chalk River of Canada (Hummel et al. 2020) have studied four different SMR technologies (HTGR, MSR, LFR and iPWR) to determine their needed EPZ size. They have used a 'limiting accident approach' based on worst case scenarios / maximum credible accidents found in literature. Without PRA available, there was no objective measure of credibility. A power level of 30 MWth and burnup approximately 23 MWd/kgU at MoC were considered. Dispersion and doses were calculated by the ADDAM code, which is part of the Canadian industry standard toolset. For some technologies, evacuation criteria were not exceeded at more than 1 km from the plant.

5.4 IAEA activities on SMR concerning EPZ size

The main starting point for IAEA work on SMR EPZ is usually that the existing basic offsite requirements like GSR Part 7 are still equally valid for SMRs as for large LWRs, but new more specific technical guidance may well be needed for SMRs. IAEA work on SMR potentially smaller EPZ for SMRs will mainly be considered in this report through the following:

- INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles)
- SMR Regulators' Forum, pilot project 3/2015 to 5/2017 (Ch. 5.4)
- TM of 2/2017: IAEA Technical meeting on next generation reactors and emergency preparedness and response (Ch. 5.5)
- CRP I31029 2018-2020: Coordinated research project (CRP) on 'Development of approaches, methodologies and criteria for determining the technical basis for emergency planning zone for small modular reactor deployment' (Ch. 5.6)

It is suggested in this report that the Finnish SAFIR2022 program would follow closely the international development and research effort in the field of SMRs. Particularly the IAEA has some projects concerning SMR technology, like INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles) and 'Common Technologies and Issues for SMRs'. Many other international activities can be found as readily well described in VTT-R-05548-16 [Hillberg et al. 2016].

Furthermore, the IAEA established an SMR regulators' forum (2015-2017 Pilot Project Report published in January 2018) for better understanding of possible future challenges in SMR regulatory discussions (see Ch. 5.4).

Related also to the SMR EPZ question, there was an IAEA meeting on emergency preparedness in Vienna in February 2017 ('IAEA Technical meeting on next generation reactors and emergency preparedness and response'). The most important discussions in the meeting are briefly included below, based on VTT attendance in the meeting.

The requirements in IAEA GSR Part 7 are basically applicable to all kinds of facilities and activities with radioactive materials. They can be used also for new generation of reactors with proper hazard assessment and graded approach (= less hazard, less need for EPR). New technical guides for the implementation of GSR Part 7 are being prepared by the IAEA. They should be applicable to all kinds of reactors, but in practice most guidance so far was for large LWRs. Dedicated technical guidance for new generation of reactors would be useful and should include:

- types of events
- selection of emergency scenarios
- sizes of the emergency planning zones and distances
    - criteria, factors affecting the choices
    - analyses used in deriving the distances
        - e.g. definition of the release source term

In many countries / regulatory regimes, an Environmental Impact Assessment (EIA) is a pre-requisite for site suitability and licensing. In 2020, the IAEA published TECDOC-1915 'Considerations for EIA for SMRs' (IAEA 2020) - not very extensive, but anyhow listing the important considerations and also trying to consider the specific issues (possibly differing from the 2014 EIA document NG-T-3.11 mainly for large NPPs) resulting from SMR new deployment possibilities. Related to the SMR EPZ question, some of the TECDOC-1915 listed considerations include:

- Footprint of the SMR site
- Modular design
- Non-electric applications

- Siting locations
- Underground construction
- Refuelling
- Waste management
- Atmospheric environment
- Terrestrial wildlife and habitat
- Human health
- Specific aspects of SMRs that influence accident assessment
- Source term for accident considerations
- Other sources of radioactive releases
- Implications of accidents on emergency planning

5.5 IAEA SMR Regulators' Forum pilot project 2015-2017

The SMR Regulators' Forum was established for member states' nuclear regulatory bodies to discuss the new regulatory issues brought by possible SMR licensing in several countries, mainly related to their innovative technology and enhanced, partly passive safety systems. These may set new challenges for example to the licensing process, graded approach, defence in depth and emergency planning. The Forum pilot project worked from March 2015 to May 2017 and gave its final report in January 2018. STUK participated in the work, particularly on defence-in-depth. Basic information and reports can be found from IAEA public website: https://www.iaea.org/topics/small-modular-reactors/smr-regulators-forum:

'The establishment of regulatory controls for this relatively new type of reactor requires focused and consistent attention. The SMR Regulators' Forum, created in March 2015, provides support by enabling discussions among Member States and other stakeholders to share SMR regulatory knowledge and experience. The Forum enhances nuclear safety by identifying and resolving common safety issues that may challenge regulatory reviews associated with SMRs and by facilitating robust and thorough regulatory decisions.'

The Forum's work is expected to consist of / result in:

- Shared regulatory experience of members preparing to license SMRs
- Capturing the good practices and methods
- Positions statements on regulatory issues
- Information to help regulators enhance regulatory frameworks
- Reports on regulatory challenges with discussion on paths forward
- Interacting with key stakeholders
- Suggestions for revisions to or new IAEA documents
- Suggestions for changes to international codes and standards; to requirements and regulatory practices

The Forum working groups (WG) during the pilot project were:

- Graded approach (GA)
- Defence-in-Depth (DiD)
- Emergency planning zone (EPZ)

Members of the Forum were Canada, China, Finland, France, Korea, Russia, Saudi-Arabia, United Kingdom and the USA. According to Stewart Magruder of IAEA (2018), the EPZ WG work was limited by the availability of SMR design information. It did not examine the public and political policies of the member states. The EPZ WG achieved the following:

- Established an understanding of each member's regulatory views on EPZ size
- Shared technical basis for EPZ size decision
- Discussed what evidence SMRs would need to present to regulators to justify smaller EPZs
- Provided, in the report, their generalized approach to determine EPZ sizes (see Fig. 6 below). The step 'Selecting events' seems to be that of a 'bounding sequence approach'.

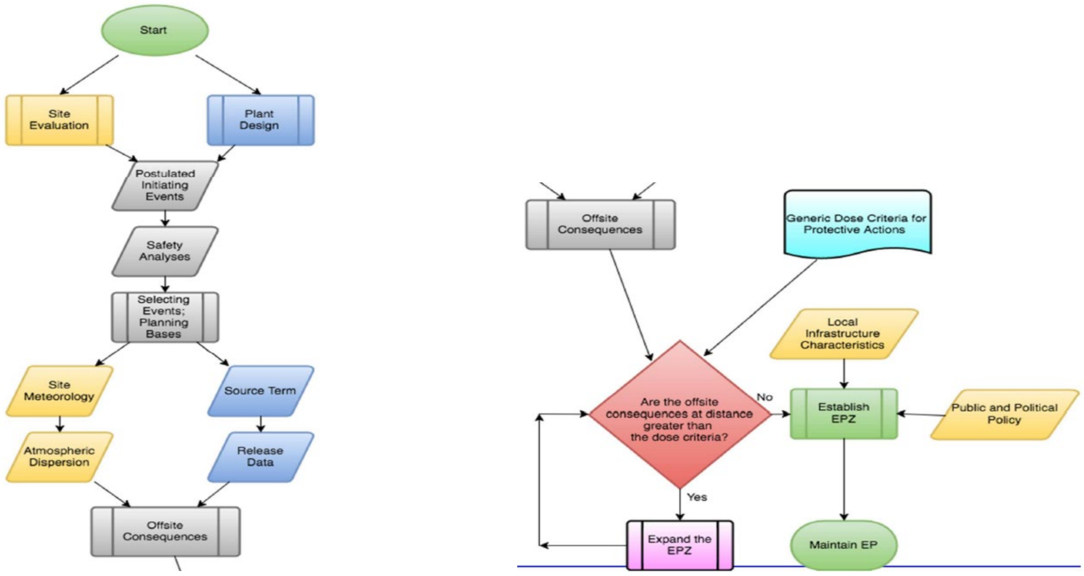


Figure 6. The IAEA Regulators' Forum EPZ WG generalized approach to determine EPZ sizes. Source of Figure: RF report (2018), Appendix IV: WG on EPZ.

The above generalized approach (Figure 6) is described in detail in the EPZ WG report, but in very concise form, the steps can be explained as follows:

- 2: site evaluation > site suitability, EPZ size determination
- 3: consideration of plant design
    - 3d: small reactor, low thermal power level
    - 3e: independent modules make large-scale offsite consequences less possible
    - 3f: containment: high-pressure, double-wall, water-immersed
    - 3g: subterranean location > smaller effective release extent
    - 3h: operating site - maintenance site: port, depot, terminal during transit
    - 3i: novel features lengthen the time between PIE and need for protective actions
- 4: PIEs (Postulated Initiating Events)
- 5: Safety analyses
    - 5b: planning bases: PSA bounds the analysis, but not the EP planning
    - 5c: source term: MST (Mechanistic Source Term) ?
    - 5d: release data
    - 5e: site meteorology
    - 5f: atmospheric dispersion modelling
- 6: determining offsite dose consequences
- 7: generic dose criteria (= intervention levels, protective action levels, protective action guides)
- 8: local effects on plant safety, impediments to emergency response; boundaries from infrastructure
- 9: public and political policy
- 10a: evaluate the plant's safety (offsite dose consequences holistically)
- 10b: compare the offsite doses to established dose criteria

The EPZ WG report (2018) and Magruder of IAEA (2018) list the conclusions / key points made by the EPZ WG:

- Different SMR designs, but coordinated EP response is needed
- EPZ scalability depends on hazard assessment, technology, design criteria, policy factors
- Refueling etc. transports: EPZ at intermediate stops & maintenance facility
- Existing IAEA safety requirements and methodology are still valid for EPZ size. For example, GSR Part 7, item 5.38: minimize severe deterministic effects, reasonably reduce stochastic effects, mitigate the accident at source; and items 4.19, 4.20, 5.38a: requirement to establish EPZ & distances.
- Possible pre-application process for siting and EPZ
- Community emergency preparedness needed, especially for small EPZ
- Novel features / technology: need mechanistic methods for EPZ size
- Different EPZ in different countries for same SMR (dose criteria, policy factors, public acceptance)
- DiD levels should be independent of each other (EPR is the 5th level, for the unexpected)
- EPZ is not a design issue (i.e. should exist regardless of the design)
- Very low probability events
- Long release duration: response in all directions needed

The country-specific sections of the EPZ WG report describe EPZ sizes & how the offsite consequences are verified for USA, Canada, France, Russia and China. In Canada, the EPZ size is flexible (not pre-determined). For future work, the WG recommends safety culture, security by design, transits, one design - one review, HFE (human factors engineering).

In November 2017 (after the pilot project), the SMR Regulators' Forum established the following new WGs:

- Licensing issues: FOAK, inspectability, testability
- Design and safety analysis: FOAK, passive safety, PEC, multi-module
- Manufacturing, commissioning and operations: from construction to manufacturing

5.6 IAEA 2017 TM on NGR emergency planning zones

In the design of SMRs, the inherent safety features are emphasized in most cases. The probability of melting of the fuel is calculated by designers to be so low that it is practically impossible. This results mainly from the smaller total power and the use of passive systems that can remove heat from the fuel without electricity and without actions from the operators. Limitation or even complete elimination of the need to prepare for offsite protective actions (mitigation of radiological consequences) has been mentioned as one of the design objectives of future NPPs. But the IAEA safety requirements in Emergency Preparedness and Response (EPR) call for taking into account also events that were not considered when designing the plant.

The IAEA arranged in Vienna in February 2017 a meeting whose objectives covered EPR and next generation NPPs: next generation design concepts and safety features & implementation of the 5th DiD (Defence in Depth) level and the IAEA safety requirements in EPR. The main reasons behind organizing the meeting were the internationally increased interest in GenX reactors, particularly including SMRs (Small Modular Reactors), their allegedly enhanced safety systems and consequently decreased probabilities and consequences of accidents, and the lack of clear or harmonized regulations of how to account for the new features in choosing the plant site and planning the required emergency response.

In any case, even with a major fission product release from molten SMR fuel, the distance of any given radiation dose level in the environment will be reduced to a fraction of that of a large power reactor, simply because of the smaller reactor core radioactive inventory. VTT has good competence in assessment of atmospheric and biospheric dispersion & the associated radiation doses, both in deterministic and probabilistic sense.

VTT (M Ilvonen) participated in the IAEA technical meeting (TM) of February 2017 on Gen4 emergency preparedness on behalf of the GENXFIN project. More information can be found in the travel report. Discussions concentrated on the possibly smaller emergency preparedness and response (EPR) distances around new types of nuclear power plants (Gen4 or SMR) and the possibility to completely do away with EPR arrangements. These achievements could be realized taking advantage of the smaller reactor core radioactive inventories and the more advanced safety features of the new plants.

For future work, four important roads forward were planned: coordinated research projects (IAEA CRPs), preparing more detailed technical guidance (possibly several EPR Series documents), discussions of plant developers & operators with EPR experts, and further meetings for follow-up.

In the official minutes of the TM, important conclusions were listed. They are still quite relevant for this review report and are reproduced here in a concise form for easy reference:

1. The DiD concept is well valid, and its 5th layer (EPR) should remain and be planned in advance. Beyond DEC events, combinations with other external events, and all different kinds of hazards should be considered.
2. New reactor designs bring more robust layers 1-4 of DiD, which leads to decreased probability of releases and smaller atmospheric source terms.
3. ‘No need for EPR’ can be a design goal, but EPR will still be needed, because of many analysis uncertainties, events not included in the design, and particularly security events.
4. Communication to the public should convince people that decreased EPR arrangements actually mean that their safety is better because of the enhanced safety features of new reactors.
5. IAEA safety standards already form a basis for design of NGRs (next generation reactors), but additional guidance will be needed, and it would be best before the NGR deployment projects even start.
6. Particularly GSR Part 7 is fully applicable also to NGR, but more detailed technical guidance specifically for NGR EPR would help.
7. Smaller core radioactive inventory, longer time to possible core melt, and lower atmospheric source term are important and quite common safety features of NGRs.
8. Public confidence will be increased if EPR considerations can be incorporated in the various stages of NGR design.
9. NGR hazard assessment has significant uncertainties, mainly because of very preliminary design status, very little operational experience, and present-day security threats (malevolent acts) with their possible implications.

5.7 IAEA 2019 TM on Advances in EPR arrangements

VTT participation in the April 2019 IAEA Technical Meeting (TM) on ‘Advances in emergency preparedness and response (EPR) technology and arrangements’ provided a general view into the international status of EPR guidelines of the IAEA, capabilities of different countries’ radiation protection authorities, and model / software development by various research institutes. The TM was not specific to SMR, but the topics are basically applicable to all kinds and all sizes of nuclear power plants. Meeting topics included operational arrangements for nuclear emergency response, newest advances in accident simulations, and plume modelling and atmospheric dispersion prediction.

Head of the IAEA Incident and Emergency Centre (IEC) Ms. Elena Buglova presented many IAEA acievements and expected future developments in EPR, including:

- GSR (General Safety Requirements) Part 7 implementation was accelerated by many activities.
- Fukushima Daiichi accident report (2015) with five Technical Volumes indicates, among other things, possible improvements in EPR arrangements.
- EPRIMS (Emergency Preparedness and Response Information Management System) helps to implement GSR Part 7 requirements.
- IRMIS (International Radiation Monitoring Information System) helps to visualize radiation monitoring data in emergencies.
- IAEA maintains a website with assessment and prognosis tools.
- USIE (Unified System for Information Exchange in Incidents and Emergencies) serves as the main emergency communications platform between Member States.
- Member States may request appraisal of their nuclear emergency preparedness level through EPREV (Emergency Preparedness Review) service of the IAEA.

Some notably interesting sessions of the meeting considered the following topics:

- Operational arrangements for nuclear emergency response
- EPR technology for first responders' use (measurement devices, personal protective equipment (PPE), emergency alarm sirens etc.)
- Advances in accident simulations
- Plume modelling and atmospheric dispersion prediction
- Advances in industry

5.8 IAEA 2019 Regional workshop on EPR for SMRs

In December 2019 VTT / the author participated in the IAEA 'Regional workshop on the principles for emergency preparedness and response for SMRs' as a lecturer. The participants were from Eastern European and Baltic countries, Russia and Azerbaijan. Notable sessions of the workshop included the following:

- Overview of SMRs and EPR (Vilar-Welter / IAEA)
- Hazard assessment (Anderson / IAEA)
- Emergency preparedness categories (Ilvonen / VTT)
- Relevant quantities and possible health effects (Ilvonen / VTT)
- Environmental characteristics of a release (Vilar-Welter / IAEA)
- Protective actions (Anderson / IAEA)
- IAEA CRP project on SMR EPZ (Vilar-Welter / IAEA)
- Risk perception (Anderson / IAEA)
- Role of the IAEA in EPR (Vilar-Welter / IAEA)
- Suggestions for further discussion in SMR EPR (Ilvonen / VTT)

The EPR arrangements for LWRs were compared with those for SMRs, through discussions, using LWR presentations as a basis. VTT lectures were about emergency classification, health effects of radiation doses, and specific considerations in arranging EPR for SMR plants. It was also discussed how the specific EPR considerations may apply to participants' countries, at national level, and about how EPR for SMRs could be developed in the future.

The impact of such a workshop is to facilitate the introduction of SMR plants in the participating countries by pointing out the special considerations that should be taken into account in their emergency preparedness planning, considering both siting and licensing. Currently there are still many problems waiting to be solved, particularly because of the vast variety of SMR plant types and sizes. It is strongly suggested that further workshops, with similar lecturing and discussion-raising, should be arranged in near future, as many member states are planning or already introducing SMR type nuclear plants as part of their energy mixture.

5.9 IAEA 2020 TM on NGR emergency preparedness & response

The September 2020 IAEA TM on ‘Next generation reactors and emergency preparedness and response’ (NGR and EPR) was essentially a continuation of the 2017 similar TM which was described above. The 2017 TM was an initial step in NGR EPR and brought together both NGR technology developers and EPR experts. The starting point of the 2020 TM can be described with the following statements:

- GRS Part 7 requirements are fully applicable to NGRs.
- Off-site EPR arrangements will still be needed for NGRs (even when a design goal was to eliminate it).
- EPR should account for the NGR hazard on a graded approach.
- Work is needed for consensus on adequate EPR (sufficient but no extra).
- Communication with the public is the key to implementation of reduced EPR.
- Additional technical guidance for NGR EPR would be helpful.
- NGR preliminary status and lack of operational experience results in uncertainty about needed EPR.
- The CRP project on SMR EPZ has moved forward with its working agenda.
- The US NRC has developed new regulations for SMR licensing.
- Transportable nuclear power plants (TNPPs) have been developed and deployed (Akademik Lomonosov at Pevek, NE Russia).

The TM consisted of the following sessions:

- Existing EPR arrangements and recent developments for Advanced / SMR / NGR
- NGR innovative design and deployment considerations that may affect EPR
- Public perception considerations for SMR / NGR
- Graded approach to EPR and perspectives related to SMR deployment
- Perspectives on EPR from embarking countries considering SMR / NGR
- Hazard assessment methodologies specific to advanced / NGR

R. de la Vega of IAEA provided some perspective on graded approach (GA) to SMR EPR. He emphasized the existence (in GSR Part 7) of Emergency Preparedness Categories (EPC), of which the ‘worst’ one is EPC I, with the potential for severe deterministic health effects off-site. SMRs may narrow the span of conceivable events, but malevolent acts could be a source of major concern. Graded approach putting SMR to lower EPC should thoroughly consider uncertainties.

L. Berthelot of IAEA presented considerations related to public acceptance of SMRs: siting close to a population center, integration into an industrial complex, modular construction, non-electric applications and waste management.

R. Kahler of the NRC started from the EPR rationale of NUREG-0396 (NRC 1978) and proceeded to explain the possible use of graded approach in emergency planning (EP) and scalable EPZ. He noted that NRC regulations actually readily provide for scalable EPZs. Since the 1980s, there were EPZ sizing studies for passive / advanced reactor designs. GA has been applied to research and test reactors, fuel fabrication, spent fuel storage etc. NRC-proposed new rule bases EPZ sizing on the risk of the facility. NRC will not try to drive the risk lower than is needed for achieving adequate protection.

CNSC (Canada regulator) presented, among other things, a list of problems facing the regulator in SMR licensing: Interpreting requirements & guidance, meaning of 'proven', acceptable level of professional judgement, needed level of evidence, results from other regulators.

Pittsburgh Technical presented their improvements at various levels of PRA, potentially resulting in right-sizing the EPZ: source term reduction by increased decontamination factor (DF), improved atmospheric dispersion models and evacuation time estimates (ETEs).

M. Ilvonen of VTT presented consequence assessment methods with special consideration of small reactors, related to nuclear district heating options in Finland.

5.10 IAEA CRP I31029: SMR EPZ methodology

Results of a previous CRP project

VTT participated in the IAEA CRP I31029 (SMR EPZ methodology) in 2018-2021. However, it is more than appropriate to mention here that a rather similar CRP project was carried out about ten years ago: I25001 (Small reactors without on-site refuelling), producing the TECDOC-1652 (IAEA 2010). The 1652 considers various aspects of the small reactors (water cooled with particulate fuel, or fast neutron spectrum with chemically inert coolant), like neutronic characteristics and development scenarios, but also emergency planning, with emphasis on the reduction of the EPZ. The EPZ study is described in the 57-page Annex I (Methodology to revise the need of relocation and evacuation measures for SMRs), which has the following main parts:

- Part 1: Review of current EPZ regulations in several Member States
- Part 2: Critical review of history of EPZ regulations (focus on 1995-2010): Previous attempts of EPZ redefinition in the USA, and resulting insights
- Part 3: Integrated methodology for EPZ redefinition, looking at frequency of exceeding given dose limit at given distance

For EPZ redefinition, the authors propose a methodology using PRA and deterministic dose evaluation. They focus on the frequency of exceeding a dose limit at a given distance, when the full spectrum of occurrences is considered. The dose limit is kept the same as before, but the EPZ distance may be reduced if the same previously used (not necessarily as an explicit number) frequency of exceeding the limit (at the old fixed EPZ distance) now would occur at a smaller distance, for an innovative SMR plant. The methodology was also published by Mancini et al. (2014) and will be described in more detail below in this report.

Latest CRP I31029 on SMR EPZ

During the IAEA TM meeting of February 2017, a new coordinated research project (CRP) was planned: I31029, Development of approaches, methodologies and criteria for determining the technical basis for emergency planning zone for small modular reactor deployment (https://www.iaea.org/projects/crp/i31029). Based on the conclusions of the meeting, it is recommended that a rigorous analysis of the EPR distances of SMRs should be made based on actual radioactive inventories, modelled DF (leak path decontamination factors) and resulting atmospheric release source terms, as well as computational assessment of doses and their comparison with international action levels for radiological countermeasures.

VTT participated in the CRP I31029, on-going in 2018-2021. In addition to VTT's own activities and contribution to the topic of the CRP, general ideas (conclusion results from all participants' relevant activities, without disclosing confidential information) are reported in this report within the EcoSMR project, for the good of Finnish NPP operators and STUK possibly encountering plans for SMR plants in the future. The main question is how far from an SMR plant the PAZ (precautionary action zone) and UPZ (urgent protective action planning zone) should reach, which is of particular importance if the SMR plants are to be used as a local source of heat for cities and industry.

The Emergency Planning Zone (EPZ) is defined as follows:

An EPZ consists of the precautionary action zone (PAZ) and the urgent protective actions planning zone (UPZ) where arrangements have been made to take precautionary and urgent protective actions in the event of a nuclear or radiological emergency to avoid or minimize severe deterministic effects off the site and to avert doses off the site in accordance with international safety standards.

The specific research objectives of I31029 are:

1. Formulate criteria for the events and technical aspects to be considered for defining emergency preparedness & response (EPR) arrangements for SMR, focusing on EPZ sizing. This should be based on the results of the research and implementation of defence-in-depth in the design of SMRs, including:
    - small power
    - smaller source term
    - increased safety margin
    - enhanced engineered safety system
    - smaller fission product release
    - consequent reduced potential for radiation exposure to population in the vicinity of the plant;

2. Develop approaches and methodologies which enable relating safety features of SMRs with the extent of offsite arrangements needed, particularly the size of EPZ, by comparing design- and site-specific technical basis to be provided by
    - SMR developers
    - nuclear regulators
    - emergency planners
    - users/utilities;

3. Provide suitable technical basis, as an input into the development of IAEA technical guidance (EPR series report) on EPR arrangements for SMRs. Also additional input into new guidance regarding source term definition and assessment could be derived on that basis, as appropriate.

The IAEA contact person for CRP I31029 is Mr. Ramon de La Vega (Emergency Preparedness Coordinator of the IAEA Incident and Emergency Centre IEC). The CRP started in January 2018 and was originally going to end by the end of 2020 (i.e. 3 years duration for the research). In the first Research Coordination Meeting (RCM1), organized in March 2018, the details for the Research Agreement signed by the IAEA and the participating research entities were established. The next meeting (RCM2) was arranged in Beijing in May 2019 as the last physically face-to-face RCM. After the COVID-19 pandemic began in early 2020, the remaining RCM meetings were arranged remotely, which somewhat prohibited informal exchange of experiences between researchers. Also, the original 3-years duration was extended up to August 2021 with one extra RCM meeting to allow more time when the pandemic is slowing down many operations in the participating institutes. The remote meetings RCM3 and RCM4 were arranged in August 2020 and July 2021. VTT participated in all the four RCMs with presentations on on-going work. After the official end of the CRP, it is expected that within appr. one year the participants will contribute to writing of an IAEA TECDOC document. From IAEA side, that work is now being coordinated by F. Stephani and H. Subki.

In the Vienna RCM1, VTT presentation outlined the steps that should be taken in order to establish rigorously determined EPZ size for an SMR plant (cf. also the steps taken by SSM 2017, described above in this report):

- Familiarization with IAEA Requirements and Guides for EPR
- Familiarization with the design of the SMR plant with related safety features
- Reactor core radioactive inventory, preferably End-of-Cycle (EoC)
- Postulated incidents and DBA accidents, also possible severe accidents
- Determine atmospheric source terms
- Weather data of a potential site, or preferably several sites
- Define dose criteria to be used for EPZs
- Choose off-site dose calculation model (e.g. MACCS, RASCAL or ARANO).
- Deterministic and probabilistic (if annual weather data available) calculations
- In the case of single dispersion condition, at least the worst and the most common dispersion condition should be used.
- If there is annual long-term weather data available, the results are weighted with the probabilities of the different weather conditions.
- Results with different source terms with site specific weather data
- Sizes of EPZ are obtained by comparing dose results to allowable dose limits.
- Comparison with EPZ sizes possibly given by plant designer
- Justified recommendation of EPZ size, based on protective measures needed at a chosen level of confidence (decision about allowed frequency to exceed limit)
- In the end, results may be compared with current large NPPs and other SMRs.

In the Beijing RCM2, VTT presented comparison calculations of ARANO, MACCS and VALMA. In the remote RCM3, VTT showed extensive ARANO and MACCS calculations for a severe accident of the Korean SMART reactor. As end results, the dose quantities listed in GSR Part 7 Tables II.1 and II.2 were predicted by MACCS as a function of distance. Then the maximum distances requiring protective actions were found by comparing with the generic dose criteria in the tables. In the remote RCM4, VTT outlined a calculation scheme for fast re-assessments of offsite doses, based on element groups, with which it is possible to directly express doses as a function of distance with a linear combination of groupwise release fractions, provided that other parameters of source term, weather and dose assessment remain fixed.

Regarding the to-be-completed TECDOC, an initial Table of Contents was proposed by EC-JRC already in RCM2. In RCM3, there was a systematic attempt to point out possible contributions from the presentations to the TECDOC. The joint closing session on the final day discussed plans for conclusion of the CRP and writing of the TECDOC. An extensive Table of Contents of five pages was produced to facilitate writing of the CRP report. In the remote RCM4 in 2021, development of a draft IAEA TECDOC was planned with the following steps in the agenda:

- Placement of each individual research in the TECDOC
- Conclusion of the CRP and workplan for developing and finalizing TECDOC
- Development of preliminary draft of TECDOC as the final report of the CRP
- Planning for a consultancy meeting in Q2/2022 to finalize the TECDOC

As of this writing (December 2021), it is still unfortunately unclear when the TECDOC will really be finalized. Work in the CRP and its aftermath has been slowed down both by several changes in IAEA staff and the COVID pandemic. In the author’s opinion, there are two very different aspects for the TECDOC, each very important:

1. General methodology / guidelines, meaning what should be calculated (releases, exposure modes, weather choice, criteria; which codes to be used; probabilistic / deterministic approaches; how to decide about EPZ when calculated results are available).

2. Sample cases, as many CRP participants did using hypothetical accidental releases from their SMR plant.

In the TECDOC, in the author's opinion, the bulk part should be about (1) and then as appendices there could be sample case results (2). It seems that generally in the CRP, most participants concentrated efforts on (2): They wanted to calculate a certain reference accident case of their certain plant type, using their chosen code. That was often without further thinking / justification on why they did in the way they did. Some examples of more general-level thinking were presentations by UK CRA-RISK, Argentina (Mr. Caputo), or JRC (De la Rosa Blul).

5.11 IAEA CRP J15002: Effective use of dose projection tools

As a background of the CRP J15002, it should be observed that dose assessment tools, like ARANO, VALMA and MACCS used at VTT, can be used for nuclear and radiological emergencies, both in the preparedness (planning) phase and in the actual emergency response phase, i.e. when a release of radioactive material has happened or is expected to happen soon. In planning, it is valuable to know what kind of radiation situations to expect in the environment (both in probabilistic sense and through chosen sample cases). In the response phase, it may be crucially important to perform protective actions even prior to the release for them to be effective. Only dose assessment models have this kind of predictive capability, in contrast to measurements. Such a radiation prediction can give information on the areas affected by the release plume and on the dose levels that can be expected, both with and without protective measures.

The CRP J15002 was initiated by IAEA Emergency Preparedness Officer Phillip Vilar Welter. However, he quit the IAEA soon after the RCM1 meeting and the coordination of the CRP is now continued by Frederic Stephani. The CRP is aiming at improving the performance of dose projection (dose assessment) tools and helping IAEA Member States (MS) to use such tools effectively. This means also recognizing the limitations of the tools, and in increasing the effectiveness of protective response actions when relying on dose assessment codes. Use of predictive simulation codes should support and speed up the decision making in emergencies, and certainly not delay any protective actions by taking additional time from the responders. Effective use of the codes means not only the actual use of the codes for calculations, but also how their use is integrated into other emergency arrangements, including the use of measurements in the affected environment. In Finland, this kind of integration is achieved in the TIUKU system of STUK. And finally, the question is not only getting model predictions and field measurements, but making the right (and timely) decisions based on them, taking into account their possible uncertainties. In a nuclear or radiological emergency, a rapid response, possibly before arrival of the radioactive cloud, may be needed. In predictive modelling, the biggest uncertainties are usually related to the source term: which nuclides, how much activity, atmospheric release starting time and duration, effective release height?

Research objectives of the CRP J15002 can be listed as follows:

1. List the uses, advantages, limitations and uncertainties of dose projection tools
2. Code benchmark analysis against real conditions in past emergencies
3. Identify factors affecting tool performance in different kinds of emergencies
4. Integration in radiation monitoring platforms, like IAEA IRMIS
5. Recommendations for improved use in emergency preparedness and response

After participant presentations, one day of the RCM1 meeting in 2020 was devoted to working in small groups in the form of structured discussions on the central topics of the CRP:

1. Use of dose projection tools
2. Data related to the use of dose projection tools

3. Source term considerations: Usually, source term represents the biggest uncertainty, and predefined source terms should be available for use in dose tools in the response phase, when time is a scarce resource.
4. Weather data; There was discussion and differing opinions on the real need for detailed & sophisticated weather data.
5. Atmospheric transport simulations: Various dose assessment tools are based on a widely different spectrum of atmospheric transport / dispersion modelling.
6. Outputs from dose tools: Sligthly different kinds of outputs are needed for different groups of recipients: experts, decision makers, or communication to the public.
7. Dose conversion factors: Factors used are not well documented in all tools.
8. Benchmarking activities: Participants emphasized the need to benchmark the actual use of dose codes, in contrast to comparing the codes themselves.
9. Future organization of the CRP

On the 1st day of RCM2 in 2021, an opening presentation was given by Mr. Frederic Stephani of IAEA IEC (Incident and Emergency Center), emphasizing the objectives of the meeting:

- Exchange information about participants' own internal programmes of work.
- Present and discuss the outputs from common CRP activities (Emergency response exercise in November 2020 and Emergency preparedness exercise in early 2021).
- Plan the coming exercises during the CRP (two emergency response exercises in December 2021, with NPP / radiological scenario) and response exercise #3 in April 2022). The level and amount of provided technical information will be complexified, as compared with the previous exercises.

According to opening remarks by Florian Baciu of IAEA, key findings of the CRP are going to be used as a basis for a new IAEA EPR (Emergency Preparedness and Response) Series document.

Frederic Stephani also presented results from a questionnaire that was sent to all participants when the CRP project was started. Questions were about the features of the dispersion & dose assessment models that the participants are using. Most typically each institute is using 3 tools. This is the case also at VTT (using ARANO, VALMA and MACCS). Models used included C3X, RASCAL 4.3, HotSpot 3.1.2, JRODOS, Nostradamus / SOPRO, and PACE. The models are of Gaussian, Lagrangian or Eulerian type, with no one type dominating. Gaussian models use traditional Pasquill stability classes, whereas other types are usually based on numerical weather data (NWP, numerical weather prediction). Most used exposure pathways are external from cloudshine and internal from inhalation.

5.12 Protection strategy: more than just EPZ size

A complete protection strategy (i.e. all the measures taken to prevent any adverse effects caused by the NPP; can be caused mainly by the radioactive nuclides in the fuel, primary circuit etc.) should basically include actions at all the 5 levels of DiD (Defence in depth, cf. Ch. 4.3 in this report). As there are no infinite resources, it is always necessarily some kind of compromise, i.e. a balance between accident prevention, accident mitigation, and protective actions.

This balance was discussed e.g. in the NEI (2013) White Paper. The appropriately sized EPZ should be the basis for a certified offsite all hazards plan, which includes defining emergency action levels (EALs), emergency drills and training, protective action strategies, and a modern public alert system. The decision on all of these is related to the decision on the size of the EPZ. The possibility of needing to expand the response efforts should also be considerd. According to NEI (2013), 'The size of the EPZ should be such that consequences from more probable, less severe accidents would not exceed the PAGs outside the EPZ, and should also provide for substantial reduction in early severe health effects in the event of less probable, more severe accidents.'

It seems that in some cases the SMR community (first and foremost, the plant providers) would like to shift the balance of protection strategy to ‘safety by design’, i.e. to make the probability of accidental radioactive releases so small that there would be practically no more need to consider the offsite protective actions. Thus there is the question, whether the optimal protection strategy could be totally different for at least some SMRs than for large plants. Ramana & Mian (2014) crystallize this idea by referring to the argument that this “safety-by-design” approach can focus on “eliminating by design the possibility for an accident to occur, rather than dealing with its consequences”. However, it has also been suspected that the safety-by-design approach might deteriorate safety culture, as it would be expected that nothing serious can happen, regardless of the actions of the employees.

Consideration of all hazards, together with all their consequences, is important. An extreme example is the Fukushima accident of 2011, where the tsunami caused the NPP accident, but actually the tsunami was responsible for almost all the acute hazard to the offsite public. Furthermore, the destruction caused by the tsunami hindered all kinds of accident mitigation and public protective measures. Of course, similar circumstances seem very unlikely in connection with SMRs, but when internal causes of accidents are designed away with, there still remain the various external causes (external hazards), which might affect all the SMR units at one site as a CCF (common cause failure). This is particularly true when the SMR is located next to industry which may have its own hazard-causing accidents.

Also the dose-averting countermeasures have radiological and non-radiological (societal, economic) consequences. For example, evacuation performed in a non-optimal way might increase the dose received in short term, if traversing the cloud or contaminated areas. Evacuation will also cause hazards of its own, resulting from traffic accidents, heart attacks from panic, etc.

5.13 International harmonization: Ideal but distant goal?

International harmonization of emergency preparedness & response (EPR) is an old topic and has progressed towards more harmonized EPR concepts. In Finland, STUK has over the years participated in Nordic co-operation and common efforts toward harmonization have been reported, like the ‘Nordic Flagbook’ guidelines and recommendations (STUK 2014). As Finland is a member in the EU, council directives like EURATOM/2013/59 are binding. STUK has also participated in WENRA (Western European Nuclear Regulators’ Association). Probably the widest promotion of international harmonization has been achieved through the IAEA.

In most countries the EPR is generally compliant with the relevant IAEA safety standards, and harmonization towards that common goal is progressing. The EPR arrangements are constantly being improved based on experience from past emergencies and various exercises and trainings. However, there is no relevant experience from the new generation of reactors. Other widely recognized problems in current EPR include harmonization across neighbouring countries’ borders, lack of financial, human and technical resources, and communication / information exchange between the involved parties and the general public.

Harmonization of emergency preparedness is a desired outcome especially because of the possible cross-boundary effects of nuclear accidents. A prevailing radiation situation could be considered acceptable in one country, but necessitating protective measures in another, even when measurements of radiation give similar values. This could even deteriorate public acceptance of nuclear energy, because such contradiction may seem suspicious, as if one country were hiding some secret knowledge of adverse effects from its citizens.

International harmonization of the NPP licensing process (LP) as a whole is a much harder goal to achieve than that of EPR actions. However, it is exactly the LP harmonization that is desired by SMR plant providers, as it would greatly facilitate the production of SMRs in large numbers and delivering them without changes to several countries. Only in that way would the full benefit of industrial serial production be realized.

Sainati et al. (2015) provide a good short discussion of potential international harmonization of the LP, in connection with more general discussion of licensing of SMRs. They extend the discussion of Ramana et al. (2013). Under ‘Regulatory harmonization and international certification’ they categorize the following groups of stakeholders:

- IAEA and other international organizations
- The nuclear industry, lobbying toward governments and regulators
- Regulatory bodies, collaborating through e.g. WENRA

Sainati et al. (2015) find it difficult to make significant progress in regulatory harmonization because of the differences in legal systems, institutional systems, LP structure and underlying principles. The NRC has a manufacturing license that does not substitute the LP but speeds it up. It may facilitate SMR industrial manufacturing, but still the actual LP applies to the final system installed on the site. They also discuss a framework similar to that used in many countries for limited thermal power and purpose (like research reactor): It could be the way forward, but difficulties arise e.g. with public acceptance.

As a conclusion, tailored licensing, shared between several nations, would be needed for wide adoption of SMRs. National states’ political commitment requires legal reforms, deeply modifying their licensing processes. A ‘design acceptance’ of an SMR that would be valid in many countries is hard to achieve and would still leave the licensing of site, organization etc. to be done.

5.14 Public acceptance of nuclear power plants

It is not self-evident how the public acceptance of an NPP would change with smaller / larger EPZ. Traditional thinking is that people don’t want to live near NPPs (‘not in my backyard’), and that a larger EPZ would make the plant more acceptable, simply by forcing it to be located in less populated areas, far away from population centers. But on the other hand, this may be exactly one of the things that make NPPs seem different from any other industry. There are several other branches of industry that may pose a hazard, like chemical plants. If an SMR would be safe enough to be accepted with small EPZ, even within a city, then maybe people’s attitudes would also gradually change. The NPPs would become more ‘part of everyday life’, routinely seen by lots of people.

Possibly difficult for broad public acceptance would be exactly the new features of SMRs, even those that were designed to maximize safety, but strange and FOAK (first-of-a-kind), subject to the claim that it is not possible to know the system before longer operating experience. Perhaps the EPZ should be larger at first and then it could be reduced, after gaining sufficient (positive) experience?

Mancini et al. (2014) state that ‘the fact that the off-site zone around NPP is treated in a special way sends an incorrect message to the public regarding the safety of NPPs and in the unlikely event of an accident could even induce among residents of the affected areas the paralyzing fatalism that is recognized to be the largest and long lasting public health problem created by the Chernobyl accident’.

Ramana et al. (2013) conclude about SMR licensing: ‘One challenge that confronts the expansion of nuclear power has been adverse public opinion. Modifying licensing requirements in order to make SMRs more economically competitive will likely impact how the public perceives these new reactor types and their deployment. If there are questions about the economic viability of SMRs, then it may be more advisable to address those through technological and manufacturing innovation.’

For the SMART design of South Korea, Ramana et al. (2013) include the following historical information on EPZ and public acceptance:
‘One reason for retaining a larger EPZ radius might be the anticipation that a reduction in the emergency planning zone might not be supported by the public. One public poll found that in the case of the APR1400, even though according to the reactor’s designers, there was a technical case to reduce its EPZ area to 700 m, only about 20 percent of the general public and about 25 percent of workers at the Kori nuclear

plant supported the idea of an EPZ area with radius below 1 km. The EPZ for the APR1400 reactor is approximately 8-10 km in radius depending on the site characteristics.'

For all nuclear plants, communication with the public is essential for acceptance. A substantial fraction of people do not understand the units or levels of radiation-related quantities, have unnecessary / irrational fear of even minute levels of radiation and anything nuclear. The IAEA EPR-NPP_PPA (2013) includes a whole section (Section 7: Communication with the public and decision makers) and Appendix III (System for placing the radiological health hazard in perspective) for these purposes. In the SMR case, communication to the public should convince people that decreased EPR arrangements actually mean that their safety is better because of the enhanced safety features of new reactors.

As an example of negative public opinion, a piece of text by 'Snake River Alliance' ('Idaho's nuclear watchdog & clean energy advocate') on NuScale proposed EPZ size is included here:

There is no justification for reducing emergency planning for new nuclear power plants. We know that any nuclear chain reaction is inherently risky. The United States has historically depended on the strategy of "defense in depth" to protect people from this risk around commercial power plants. This means there are multiple independent and redundant defenses to try to prevent accidents or lessen their harm. Shrinking the EPZ is absolutely counter to the "defense in depth" strategy — particularly when applied to new and untested reactor designs.

Proposed rule change: For the sake of reducing the regulatory burden on developers, the NRC is now considering NuScale's proposal to shrink the emergency planning zones, not just for NuScale, but for all SMRs and other new reactors, too. The change would reduce NuScale's EPZ all the way down to just the land within the 40-acre facility.

This is a terrible idea! Shrinking the EPZ can only be justified if one accepts the nuclear industry's claims that new reactors will be safer. Those claims are unprovable since these new reactors don't exist yet and have not been tested. All Idahoans have a right to know what would happen in case of an accident at NuScale.

An accident in a single small reactor can have big results. SMR developers plan to concentrate a number of small reactors near one another. NuScale, for instance, plans to build 12 of its 50-MW reactors in a single buried chamber. A 600-MW nuclear reactor is not small, and the potential effects of an accident are not either. Public safety demands far more than a single 40-acre emergency planning zone for an array of 12 reactors.

Public acceptance of nuclear in Finland is quite good according to a 2019 energy attitudes survey: 47 % for increasing nuclear capacity (particularly among 18-25 year olds), and 23 % for decreasing it. Even some prominent Finnish Green Party politicians are pro-nuclear, a trend which has become more pronounced in 2021. More cross-society discussion may be needed before wider acceptance. Generally, higher level of education may contribute to acceptance of NPPs, as it leads people to trust science / experts and also trust the decision-makers. To quote Professor J. Leppänen of VTT, 'All SMR concepts aim to benefit from simplified design. When you have simpler design and fewer moving parts, that means it's easier to demonstrate safety and that helps with public acceptance.' Stronger safety may be achievable with less emergency preparedness than before. In fact, a large EPZ size might even cause more public fear than confidence. However, spent fuel & transports are always also part of the acceptance problem.

Whatever the solution to the SMR EPZ size question is going to be in the technical sense, final decisions will be affected by public perception of risk: Will people in a city trust SMR safety? Unfamiliarity, uncertainties & non-voluntariness contribute to perception of risk. For example, when comparing hydropower vs. nuclear power, even with more deaths caused by dam failures than by severe reactor accidents, for most people 'dams are comfortably familiar' (Waller & Covello 1984).

5.15 SMR-specific considerations in EPZ determination

SMR new features are generally designed to maximize safety, but ‘provable safety’ may be more difficult than for traditional large LWRs, simply because the enhanced safety features have many unknowns. In some cases, new features may also bring new risks that should be considered (e.g. multi-module plant, common control room).

Magruder (2017) and Vilar-Welter (2018) of IAEA SMR Regulators’ Forum & CRP I31029 mention passive systems, slow progression of accidents, multi-module plants, control room, human factors, physical security at new sites, low probability events, the intention to create ‘safety-by-design’, and the non-linear effect of source term on the size of the EPZ.

A listing of (generally) SMR-specific, or requiring new thinking in the case of SMR, considerations in SMR EPZ determination was compiled by the author of this report and presented in the IAEA CRP meeting as well as in a Safir2018 meeting (5/2018):

General open questions in SMR EPZ determination:

- How to use probabilistic vs. deterministic calculations (for source term, or off-site doses)
    - E.g. PSA and MELCOR use
- Conservative vs. best estimate assumptions
    - 2017 NuScale calculations in GENXFIN were very conservative
- To what extent, need for sensitivity analyses
- Detailed modelling of everything vs. simpler models / estimates / expert judgement
    - Complete detail is impossible and may even lead to pitfalls
- Local conditions vs. generally acceptable rule
    - External events, weather conditions, environment, population
    - SMRs pre-fabricated; and floating plants exist
- First-of-a-kind (FOAK) plants
    - Too little statistics of components functioning?
- Safety features: inherent, passive, active
    - Inherent: Physical impossibility, or indisputable property
    - Engineered systems:
        - Passive (may still malfunction)
        - Active (best previous understanding?)
- Leaving out very improbable sequences / very small releases
- Combining sequences in PSA 1 > 2 > 3

Level 1 PSA open questions:

- Very low probability / extreme events
    - Actual value of probability very uncertain?
    - Frequency threshold?
    - PEC (practically eliminated sequences)
- Multi-unit case:
    - Probability that any one of the units has accident
    - CCF (common cause failure)
    - Accident can even progress to other units?
- External events, including hostile?
    - Many SMR-planning countries are not the most peaceful
- Note: Some designs (MSR) readily have the core in molten form
    - Definition of core damage?
- Severe accident simulation tools suitable for SMR?

Level 3 PSA open questions (cf. also Chapter 10.2 for more details):

- EPR (Emergency Preparedness and Response) as independent (5.) DiD level, regardless of previous barriers?

  - However good barriers: Still after design, probabilities may be affected by quality in construction and operation.
- EPR should exist for the unexpected / unconsidered
- How to reach public acceptance for EPZ
  - Also national differences vs. international harmonization
- Simple scaling by power level? ‘Inventory = damage potential’
  - Even now, we are not thinking that the whole inventory could be blown into the air.
  - Scaling by power level would simply shift down the ‘dose as function of distance’ curve.
  - Find the distances with same doses as now 5 km / 20 km for big NPP.
- Criterion to look at: Dose (=consequence) only, or risk (= dose x probability) ?
- No urgent actions needed if releases are provably always delayed
- Possible SMR-specific chemical compositions of release (NaI ?)
- MST (mechanistic source term) modelling needed?
- How many years of weather data?
- Uncertainties of atmospheric dispersion
- Near-field dispersion more important for SMRs
- Cloudshine in near-field from actual integration, not DC (dose conversion) factor
- Doses to foetus & thyroid may be decisive for EPZ size
- Dose limit reference levels, frequency-related dose levels
- Choose 95 % / 99.5 % / etc. confidence level from distribution of doses?
- Vulnerable groups (school, hospital, elderly home)
- Local community boundaries

There is clearly an important role for PSA in supporting decision making about EPR requirements. However, for new reactors PSA models could have a lot of uncertainty, because there is not much operational experience on them.

6. Potential methods of EPZ determination

Vilar-Welter (2018), Emergency Preparedness Officer of the IAEA, points out the high-level steps with key considerations that should be taken into account to determine the right-sized EPZ for SMR or any other NPP:

1. Hazard assessment, also very low probability & BDBA
2. Atmospheric source term, including timing
3. Public offsite doses
4. Generic criteria (doses) for response actions
5. Effectiveness of response actions
6. Available resources
7. Integration into overall protection strategy
8. Adaptation to national & local circumstances
9. Optimization

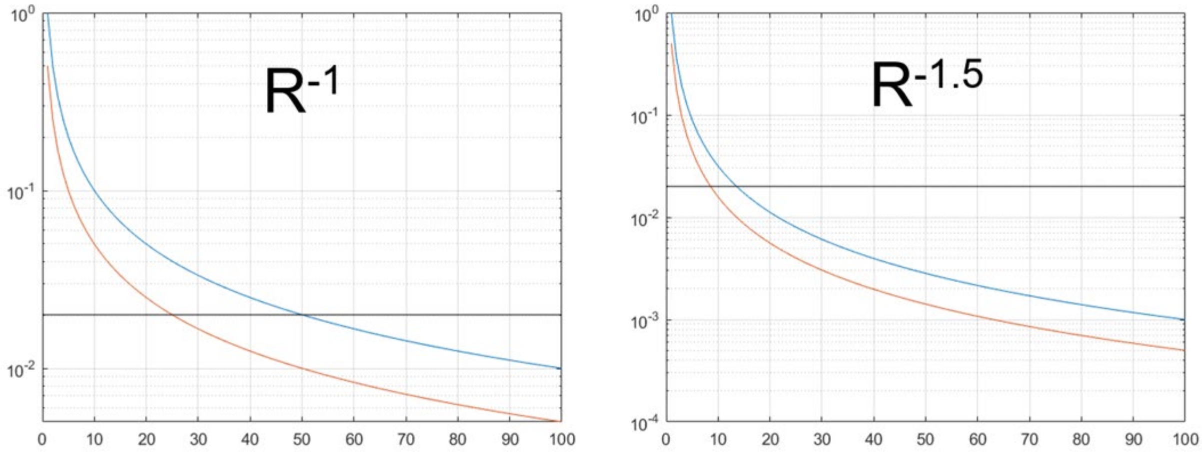
Vilar-Welter (2018) sees 1-5 and 7 of above list suitable for IAEA developments, and says that 4-6 and 8-9 should always be developed at national level. This Chapter (Potential methods) concentrates, in brief overview sense, on 1-3, i.e. some specific considerations in selecting initiating events and how to perform the dose assessment.

6.1 Simple scaling by thermal power

A lot of previous knowledge already exists particularly about offsite doses as a function of distance resulting from LWR (light water reactor) accidents. Such studies have been performed for licensing by plant providers and electric utilities, and by research institutes. A notable historical achievement with full-scope PSA for 5 US plants was the WASH-1400 reactor safety study by MIT in 1975.

There are also simple ‘rule-of-the-thumb’ formulas expressing the typical decrease of dose (at a certain time point) when the distance from the release source increases. In more advanced calculations, the dispersion of radionuclides in the atmosphere takes place horizontally, covering an increasing area, and up to a certain point also vertically (usually up to the mixing layer height). At the same time, the cloud is scavenged by radioactive chain decay and deposition processes (both depending on the nuclide content). In practice, when the dose vs. distance r curve from a sophisticated calculation is fitted with a simple function of the form $1/r^a$, it appears that the constant parameter a would usually be something between 1 and 2; often around 1.6, but exact value depends on the nuclide content and many other parameters of the assessment.

It can easily be seen that the effect on distances, when the (whole) source term and consequently all the doses are scaled down, will depend on the exponent a. For a simple pure inverse relation of the form 1/r we see that halving all the doses will also halve the maximum distance where a dose limit is exceeded, and so on for any scaling factor. But for $a > 1$ halving doses will not halve the distances (at certain fixed dose criteria). Usually dose decrease with distance is more rapid than 1/r (inverse of distance). In the figures here below, the situation is exemplified with 1/r relation (left-hand side) and $1/r^{1.5}$ (right-hand side). Blue curve represents original ‘dose vs. distance’, whereas the red curve is half that dose everywhere. Clearly with $1/r^{1.5}$, the ‘EPZ distance decrease factor’ is less than the dose (or whole source term) decrease factor.



This kind of existing knowledge offers a very simple approximation of dose vs. distance for LWR type SMRs: The existing dose curve can be multiplied by the ratio of thermal powers, which will scale it down by a constant multiplication factor, e.g. 200 MW / 4300 MW = 0.0465 to compare NuScale SMR module and Olkiluoto-3 EPR reactor. This kind of scaling is probably overly conservative, because we might assume that NuScale safety systems offer even better scrubbing and delay times before release than the EPR plant. So the conservatism comes by forgetting about any possibly more advanced safety features.

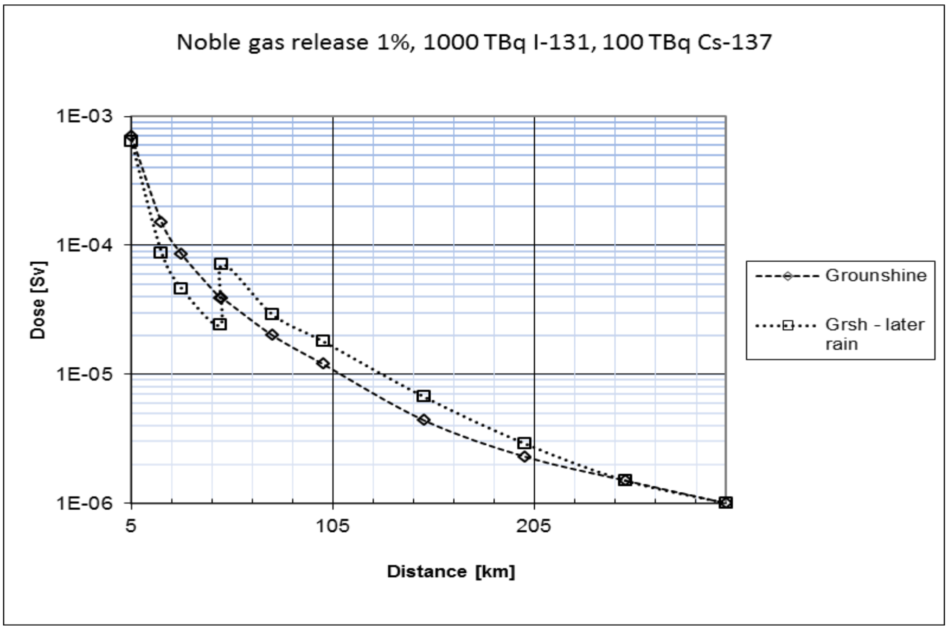


Figure 7. A dose vs. distance curve (shown in this case for distances 5-300 km, large LWR), calculated using ARANO in the COOLOCE_E project by J Rossi (VTT-R-00432-15). The release corresponds to the severe accident limit in Finland (100 TBq Cs-137). Several years' weather data was used and the 95 % fractile is shown (dose that was exceeded with 5 % probability). However, this was a partly imaginary case where the rain occurrence is delayed to begin not at the release point but only at 50 km from the power plant.

Fig. 7 shows a sample dose vs. distance curve for a large LWR, to give an idea of how dose is expected to decrease with distance in a relatively sophisticated assessment. Note the logarithmic vertical axis. The same kind of information in another form (acceptable plant size vs. EPZ radius, when extending up to a limiting dose) is shown in Fig. 8. This graphic also shows how the smaller size and advanced safety systems of SMR could theoretically affect the results.

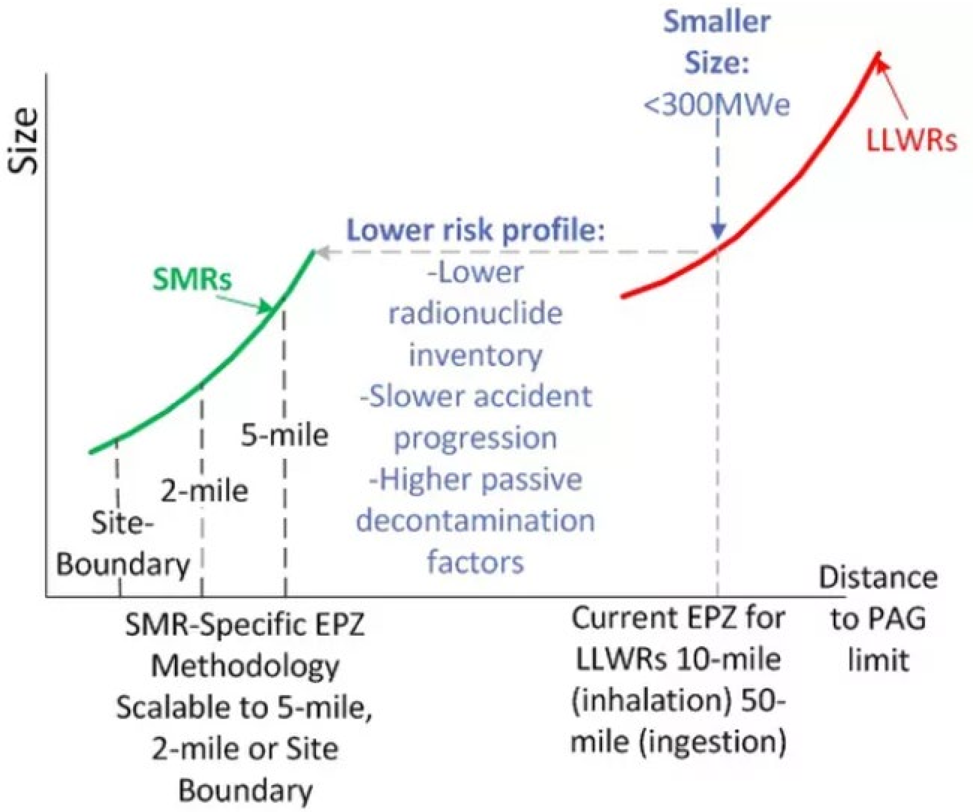


Figure 8. Size of NPP that can be sited inside an EPZ with the given radius (basic principle only, without units). Size only (meaning smaller radioactive inventory) would effectively continue the same curve further downwards, but more advanced safety features may additionally shift it to the left to smaller EPZ radii. Source of Figure: Pittsburgh Technical, www.pit-tech.com.

If moving from simple scaling to actual calculations of offsite doses, it is possible to choose to calculate some representative accident scenarios ('bounding sequences'), or doing considerably more work, to perform a complete PSA study from level 1 to level 3.

6.2 Provable exclusion of large releases

IAEA SSR-2/1-Rev. 1 (Safety of Nuclear Power Plants: Design) states, in connection with DEC (Requirement 20), that 'the design shall be such that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is practically eliminated'. The concepts 'early' and 'large' may be explained in the following way:

- Early release: Offsite emergency measures are needed, but there is not enough time to perform them.
- Large release: Need for public protective measures that cannot be limited in area or time.

Possibility of provable complete exclusion of any possibility of severe accident releases is claimed by some SMR plant providers based on physical characteristics & phenomena of the reactor. For the claim, all potential severe accident phenomena, with all possible cliff-edge effects, should be identified. If the claim holds, we could theoretically get rid of the EPZ and rely on the site boundary only. However, it is always possible to come up with external events that could blow up the plant. The IAEA SMR Regulators' Forum, DiD WG states that an exclusion should never be used to justify the omission of a complete DiD level. The situation that some accident sequence can be completely excluded from consideration may be called Practical Elimination Condition (PEC).

In some SMR types, notably some high-temperature gas-cooled reactors (HTGR, e.g. the Chinese HTR-PM), there are design characteristics that allegedly exclude the possibility of core degradation and large fission product releases completely. Among others, Reitsma (2018), Nuclear Power Technology Development Section of the IAEA, describes the basis for such claims. Core meltdown is not possible, because decay heat can be removed by natural means only. Containment of fission products in a coated ceramic fuel particle up to the temperature of 2100 K (beyond maximum possible accident temperature) should completely exclude large releases. Fuel failure cannot spread to other particles and does not affect the coolability of the fuel. Absence of cliff-edge effects makes the reactor much more forgiving than other types, and the EPZ could allegedly be set as small as site boundary.

Also, a heating-only reactor (like the Chinese District Heating Reactor DHR-400) can be a very special case, because moderate temperatures are used and no high pressures are needed, like in the case of steam turbines. Low (even atmospheric) pressure coolant can be kept in relatively strong vessels without too much expense and can be added more easily if leaked out. These conditions could be taken into account when deciding the extent of the EPZ.

Furthermore, with other reactor types there may be claims that even a molten core will not cause atmospheric releases under any circumstances. In practice, there will be some small probability of releases.

The Indian SMR design AHWR (Advanced Heavy Water Reactor), described e.g. by Ramana (2013), is claimed to be 'so safe that even in the worst of accidents, there would be no long-term impact on the people near the plant and thus nuclear plants can be constructed anywhere, even in the heart of densely populated cities'. Considering the Indian regulator AERB, there are expectations that 'the AERB will revise the siting criteria for the AHWR, because it has a multitude of advanced, passive safety features that rule out any impact in the public domain'.

6.3 Practically eliminated / impossible sequences

Some considerations specific to certain SMR types about e.g. the 'absolute', physical prevention of melting of fuel were given above in Ch. 6.2. More general and formal discussion of events / accident sequences that can be excluded from further analysis with justification is included here.

IAEA SSR-2/1-Rev. 1 (Safety of Nuclear Power Plants: Design) defines, in connection with DiD, DEC (Requirement 20) and Fuel handling and storage systems, a 'practically eliminated possibility' with the following words:

'The possibility of certain conditions arising may be considered to have been 'practically eliminated' if it would be physically impossible for the conditions to arise or if these conditions
could be considered with a high level of confidence to be extremely unlikely to arise.'

Thus a Practically Eliminated Condition (PEC) can be one which cannot come into existence because of e.g. pure physical impossibility. Numerical values of the 'high level of confidence' or 'extremely unlikely' are not given. Choice of a low-frequency 'screening out' threshold is very difficult. Leaving something out of further analysis should only be done with extreme care: 'When in doubt, don't screen it out'.

The concept of physical impossibility here does not necessarily mean one single physical phenomenon, like fuel not being able to melt in the reachable range of temperatures. It can include engineered systems, or natural phenomena (passive / inherent), or even use of undoubtedly sufficient checking procedures (testing / inspections / surveillance). FOAK designs and passive safety systems are typically problematic. Possibly still harder to prove is the other statement, that the frequency of a certain condition is indeed extremely low. This is particularly true for initiating events resulting from external hazards. Uncertainties and sensitivity of the results with respect to assumptions made should be analyzed, but uncertainty analysis of severe accidents is not mature.

IAEA-TECDOC-626 ‘Safety related terms for advanced nuclear plants’ defines an inherent safety characteristic as a *fundamental property of a design concept that results from the basic choices in the materials used or in the other aspects of design which assures that a particular potential hazard can not become a safety concern in any way*. When a hazard has not been eliminated, specifically engineered safety systems, structures or components are used to supplement the inherent features.

The IAEA SMR Regulators’ Forum, DiD WG concludes that rules for excluding identified initiating events from the design are not established for SMRs. They also state that despite SMR designer efforts oriented towards SA prevention based on DiD levels 1, 2 and 3, independent features for severe accident mitigation (DiD level 4) should be included in the design of SMRs in order to ensure the successive levels of DiD remain. The DiD WG lists the following important documents referring to PEC:

- WENRA, Safety of new NPP design: Core melt accidents with early or large releases must be practically eliminated.
- IAEA SSR-2/1 (Rev. 1): DEC with possibly significant radioactive releases must be practically eliminated.
- IAEA TECDOC-1791 (Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants): Ch. 7, The concept of practical elimination
- OECD/NEA/CNRA, Implementation of defence in depth in nuclear power plants: Practical elimination of significant radioactive releases should be addressed in the design of new plants (both prevention and mitigation).

The DiD WG expresses in their report the common position that ‘The practical elimination concept should not be used to justify omission of a complete DiD level. For example, the concept should not be used to justify absence of severe accident management arrangements and capabilities that are expected at DiD level 4 or absence of offsite emergency response at level 5.

The DiD WG requires from designers that ‘If some initiating events are considered to be excluded by SMR designers, without any safety features to mitigate their consequences, sufficient provisions (e.g., design, fabrication and operation) shall be implemented and duly justified. For the IAEA, the WG recommends that ‘Criteria for exclusion of events should be established.’

6.4 Full-scope PSA vs. selected scenarios

Traditional deterministic safety analyses mean selecting certain accident sequences (specific representative scenarios), hopefully including the worst or ‘bounding sequence’ and calculating the offsite doses with their resulting atmospheric releases. Full-scope PSA (Probabilistic Safety Analysis), with level 3 PSA, requires orders of magnitude more work (example: up to 200 man-months mentioned in IAEA 050-P-4 for level 1 only). It can produce probability distributions of offsite doses, and also useful information, like the ‘dominant sequence’ (one that brings most contribution to the offsite risk). Even with PSA (probabilistic in nature), some deterministic analyses are usually needed to generate more detailed information according to the actual phenomena.

The IAEA Regulators’ Forum DiD WG included in its report a recommendation that PSA should be used for SMRs:

‘Even if the design relies firstly on deterministic bases, probabilistic safety assessments could bring about many insights about the safety of SMRs, as they have for large reactors. Experience gained from the use of PSAs has revealed that, even when carried out from the very early design stage of a reactor, PSAs are very beneficial to evaluate the application of DiD, to check that the DiD principles have been properly applied and to identify potential weak points in the design not revealed by deterministic analyses.

Indeed, relying on a systematic investigation and assessment of a large set of initiating events and sequences, PSA results help identify the dominant contributors to the risk and thus to point out key safety issues. In particular, PSA results reflect the reliability of the features implemented at each of the DiD levels and the independence of the DiD levels. They are also useful to check the sufficiency of the redundant and diversified features implemented and to verify that the risks of common cause failures are limited. PSAs could also contribute to the identification of the postulated initiating events and of the set of design extension conditions to be considered in the design.

For all these reasons, the WG position is that for SMRs, PSAs should be used to complement the deterministic approach on which the design first relies – just as they are for large reactors.’

According to discussions in November 2018, STUK may not trust PRA level 3 in defining the SMR EPZ, because the process (PRA1-2-3) is too complicated to be even thoroughly checked.

NEI (2013) lists several risks that are difficult to quantify or not fully addressed in the PRA:
‘Security events, collateral damage, potential common cause effects on modules having common or shared systems, co-location, organizational performance, aging effects, factors affecting operations (e.g., shift staffing, training and procedures, use of new I&C systems, and lack of operating experience), errors of commission, design faults, risks treated outside the PRA due to an endorsed standard not being available, and concurrent hazards.’

One alternative to determine the EPR criteria is based on the so-called ‘maximum credible accident’: Worst case (deterministic) radioactive source term from containment into the atmosphere should be calculated and used in consequence analysis.

A more laborious, but also more mechanistic (‘first principles’) approach is performing the complete plant-specific PSA/PRA (probabilistic safety / risk analysis), including also deterministic investigations of phenomena, that would have the following phases (cf. Chapter 7.1 for more details):

- Level 1 (CDF, core damage frequencies):
    - Transient scenarios leading to core damage
- Level 2 (atmospheric releases with their defining parameters and frequencies):
    - Inventory, release from fuel, release through containment barriers
- Level 3: Offsite doses, possibly also health effects, with their frequencies:
    - Use real site-specific weather data of several years

(Note: PSA could also be used during the actual emergency response, where it can help to point out the spectrum of possible consequences particularly in the early phase, when information on the accident is very uncertain.)

CDF and LERF

From safety point of view, important frequencies from a plant PSA study are CDF (Core Damage Frequency) and LERF (Large Early Release Frequency). Their definitions according to NRC and IAEA glossaries are the following. The modifiers ‘large’ and ‘early’ do not have numerical limits in those definitions.

- CDF = An expression of the likelihood that, given the way a reactor is designed and operated, an accident could cause the fuel in the reactor to be damaged.

- Large release = A release of radioactive material for which off-site protective actions that are limited in terms of times and areas of application are insufficient for protecting people and the environment.
- Early release = A release of radioactive material for which off-site protective actions are necessary but are unlikely to be fully effective in due time.

Khatib-Rahbar et al. (2004) have considered the definition of LERF in more detail: The definition of a 'large' release and the associated time (early / late) should be evaluated from the regulatory perspective and the potential implication of severe accidents (SA), as CDF and LERF have been considered as suitable metrics for making risk-informed regulatory decisions.

Khatib-Rahbar et al. (2004) provided a literature survey of alternative definitions of large release for estimation of the large release frequency following postulated SA:

1. Any releases that occur because of SA that would entail an early containment failure (including containment isolation failure)
2. (1) above and containment bypass conditions
3. Any release exceeding specific thresholds in terms of fractional releases and timing
4. Any release occurring because of SA that would entail 10% or more of the initial core inventory of iodine
5. Any release exceeding specific thresholds in terms of the activity associated with the release from the containment
6. Any release resulting in one or more early fatalities offsite

NEI (2013) require, under 'Cumulative plant risk design objectives quantified by PRA', for SMR internal & external events that the total mean CDF < 1E-5 per plant year, and mean LERF < 1E-6 per plant year (note: not reactor-year).

PSA level 3

The necessary phases of a PSA level 3 procedure (based on level 2 results) are briefly outlined here. The main input needed for SMR EPZ calculations is the source term: what is the exact inventory of the specific type of reactor and what fractions of the nuclides will be released into the environment, what will be the effective release height, and also the expected temporal behaviour of release. As there are endlessly many different combinations, usually only a few different representative source terms, with significant probabilities from PSA level 2, can be calculated. With both in-house and NRC dose assessment codes, VTT can perform licensing safety analyses of SMRs if the compliance with dose limits has to be shown.

With the objective to evaluate and determine EPZs for an SMR plant, which has particular design and safety features, the postulated source terms have to be determined. Then, applying international safety criteria, the distances for the EPZs can be obtained. Atmospheric dispersion conditions affect significantly the doses, thus the limiting conditions should be carefully selected. Weather data may be based on an individual specified (worst) condition or the measured annual data. In the latter case, a probabilistic approach of dose results can be adopted. Exposure pathways should include all the relevant pathways, at least such as external radiation from the plume, inhalation, external radiation from the deposition on the ground, and ingestion pathways. Local environmental data may be used including shielding factors and diet. The main desired outcome is to prepare recommendations, with proper justification, on the emergency planning and response for the plant. This includes EPZs based on the international safety standards.

If an as-objective-as-possible methodology for proper sizing of the emergency preparedness zones around an SMR type of nuclear power plant is desired, as detailed and realistic data (measured or estimated) as possible should be used in calculations including e.g. postulated accidents, atmospheric releases (nuclide-by-nuclide released activities) and site specific environmental and weather data (if available). Verified and qualified calculation models should be used. When the determination of the size of EPZs is the main objective, they should be based on the relevant national (STUK) and international (IAEA) criteria. Smaller atmospheric source terms are due to smaller core radioactive inventory and enhanced safety systems.

Objective, rigorous calculations can show the actual extent of offsite arrangements needed. This can be compared with that provided by plant vendor and the guidance presently provided by nuclear regulatory authorities.

The essential pre-requisites that should be well known / in use for research work on EPZ size are:

- Site areas and EPZ sizes claimed / recommended by SMR designers
- Design of the investigated SMR plant (if one specific)
- IAEA Requirements and Guides for EPR (emergency preparedness & response)
- Potential codes for determining the releases to atmosphere and their frequencies
- Potential codes for offsite dose assessment, possibly including health effects

6.5 Site boundary vs. minimum size of EPZ

As described above in this report, some SMR vendors would like to completely eliminate offsite emergency planning and EPZ. Practically, all such planning would then be reduced to the actual site area (exclusion area boundary, EAB). However, there is the question, if there should be some minimum EPZ size in any case (conservative approach), completely regardless of how safe the plant was designed.

Mancini et al. (2014) claim that their methodology which combines probabilistic, deterministic and risk management methods, supports licensing with reduced emergency planning requirements. It can be used to devise technical criteria which would be necessary if the emergency planning is to be eliminated or reduced.

The IAEA Regulators' Forum (2018) EPZ WG concludes:
'According to existing IAEA Safety Standards, it would not be appropriate to consider EPZ&D as a design issue (i.e. as being related / influenced by the design safety).'

Likewise, the DiD WG concluded:
'For DiD level 5, the DiD WG is in agreement with the NEA statement that, no matter how much other levels may be strengthened, effective emergency arrangements and other responses are essential to cover the unexpected.'

Philip Vilar-Welter of IAEA (2018) lists among problematic approaches in SMR EPZ determination:

- The misunderstanding that the probability of an accident has an impact on the size of the affected areas.
- The assumption that the new design and new technologies eliminate the need to contemplate accidents not considered in the design, which clearly contradicts existing IAEA Safety Standards (i.e. Requirement 4 of GSR Part 7).

Ramana et al. (2013) write that 'in the early 1990s, the NRC was considering advanced LWR designs with passive safety features. At that time, NRC had noted that the promulgation of emergency planning requirements following the TMI-2 accident was not premised on any specific assumptions about severe accident probability. Hence, as a policy matter, it may be that even very low calculated probability values should not be considered a sufficient basis for changes to emergency planning requirements.'

History shows that severe accidents at NPPs have happened, and by extrapolation, they can still happen in the future, regardless of all the improvements made. As a general conclusion, without referring to assessment of offsite doses, it can be said that EPR (emergency preparedness & response) will probably always be needed, however good the design of the reactor is, simply because the potential release source term is always present in the form of the core radioactive inventory. Even with a good design, military or terrorist action and other very low probability or beyond design events are always possible to damage the reactor. Inherently safe, passive and robust systems (leading to no need for EPR), particularly by improving levels 3 and 4 (control of DBAs / control of severe conditions) of DiD, can be a design goal, but still appropriate offsite emergency arrangements should be available. Their extent is a subject of intense debate between reactor developers and EPR people.

7. Full-scope PSA for rigorous EPZ size determination

The main suggestion of this report is that the only really definitive and rigorous way to determine appropriate ('right-sized') emergency planning zones (EPZ) for SMR or any other type of plant is the laborious full-scope (i.e. 3-level) probabilistic study (PSA).

In Canada (see Ch. 5.3), as described in the IAEA SMR Regulators' Forum (2018) report, PSA is an important part of the process of flexible determination of EPZ sizes.

7.1 Introduction to full-scope PSA: From PIEs to offsite doses

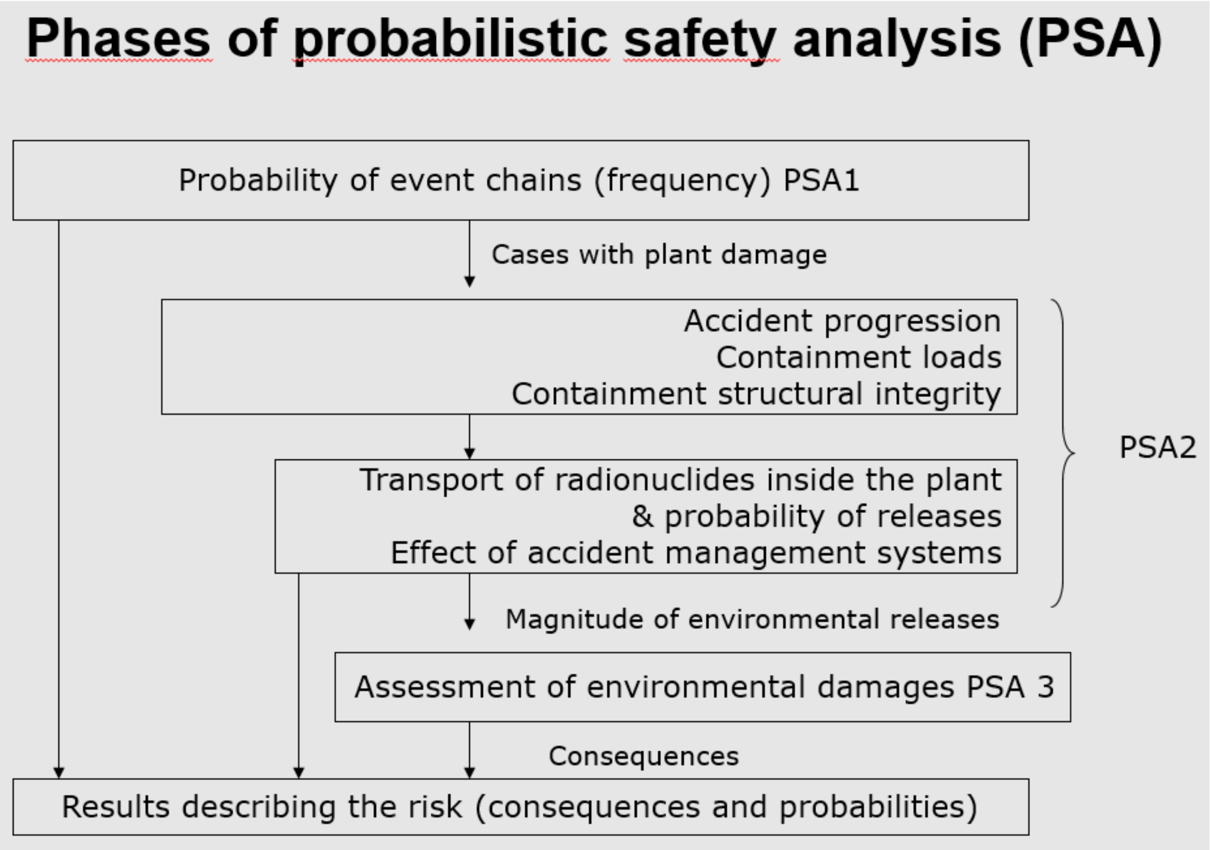
For probabilistic assessments, several different abbreviations have appeared: PSA, PRA or PCA. According to Ilkka Karanta of VTT, they can be explained and distinguished from each other in the following way:

- PRA (Probabilistic Risk Assessment): Risk is a wide, general concept, which means the realization of a damaging / negative consequence, together with its probability to be realized (risk = consequence x probability).
- PSA (Probabilistic Safety Assessment): Safety means the avoiding of human health or life-threatening risks, so PSA is a part of the more general PRA.
- PCA (Probabilistic Consequence Assessment): Consequence is a neutral concept, compared to safety or risk. Consequences could be positive or negative, or not cause any particular benefit or damage.

PSA is conventionally divided in 3 levels:

- Level 1 PSA considers the event trees (succession of events, like existence of fault, or a decision made) that start from a PIE (postulated initiating event) and end at fuel damage, e.g. loss of structure of fuel located in a reactor core by melting. The main result from PSA1 are the core damage frequencies (CDF) resulting from the various PIEs. So understanding the PSA helps to understand how credible an allegedly low CDF really is. Even for SMR, a complete list and description of both DBA and DEC postulated accidents should be available to investigate.
- Level 2 PSA studies the functioning of the containment and its systems. The event trees may contain branchings like whether the scrubber of FCVS (filtered containment venting system) works properly or not. The main result from PSA2 are potential atmospheric release source terms (cf. Ch. 8) and their frequencies of occurrence. One particularly well-known result is the large early release frequency (LERF).
- Level 3 PSA calculates offsite consequences from the releases, e.g. fatalities, other health effects, economic losses and even other, 'societal' effects. At this level the branching of possible events results from the various site-specific weather conditions (for a simple dispersion model, these are: wind speed & direction, stability class and rain).

The relations of the three levels of PSA are shown in the figure below.



Phases of probabilistic safety analysis (PSA)

Probability of event chains (frequency) PSA1

Cases with plant damage

Accident progression
Containment loads
Containment structural integrity

Transport of radionuclides inside the plant
& probability of releases
Effect of accident management systems

PSA2

Magnitude of environmental releases

Assessment of environmental damages PSA 3

Consequences

Results describing the risk (consequences and probabilities)

The ‘ideal’ probabilistic (PSA1-2-3) procedure to determine right-sized EPR zones from SMR potential accident radiological consequences, producing a plant- and site-specific result, is briefly described by the following basic steps:

- Core radioactive inventory (expected average burnup, or EoC)
    - Origen, or Serpent (3D Monte Carlo neutronics code by VTT)
- Fuel handling accidents (FHA), Spent fuel pool (SFP) and transports must be considered
- Postulated initiating events (PIEs) leading to accidents
- Internal & external events; Including malevolent actions might be more important for SMR plants, expected to be located nearer population centers.
- Accounting for collocation (multiple units); common cause, or spreading of accident conditions by effects on nearby modules
- One should perform a complete plant- and site-specific PSA study, but including also deterministic investigations of phenomena
- PSA level 1 (from PIEs to core damage frequencies CDF):
    - Transient scenarios leading to core damage
- Possibility of severe accidents (even with complete core melt) should be considered, if it cannot be completely ruled out, even when it has a very low probability.
    - Cf. Euratom Safety Directives (2013/59, 2009/71, 2014/87)
- PSA level 2 (atmospheric radionuclide releases with their frequencies):
    - Inventory, release from fuel, release through containment barriers and systems
- Estimates of the atmospheric release source terms (nuclides Bq, height, timing, temporal behaviour), with frequencies of occurrence
    - Event chain, frequency and basic parameters from PSA study
    - Detailed computational assessment using integral SA code (e.g. MELCOR, ASTEC, MAAP)
    - (Approximate assessment possible by ‘expert judgement’ starting from core inventory and decontamination factors DF)

- Among all the cases, it may be of some interest to find the 'worst case' radioactive source term from containment into the atmosphere. Such a definition is not straightforward, as it depends on nuclide composition, activities, timing etc. However, a large spectrum of accidents / releases should be considered.
    - To avoid too many source terms (to be calculated in practice), it is customary to join very similar releases into one representative source term with the total sum of frequencies.
- Consideration of site-specific conditions, like weather, surrounding terrain & environment, population, traffic connections (for evacuation) etc.
- Selecting and acquiring representative weather data (several years, hourly)
    - Data from a 3D NWP model is best. However, simpler models with simpler weather data (like one-point data from a mast) run faster, and errors in e.g. straight vs. winding trajectory will usually cancel out in a long-time probabilistic study.
- PSA level 3: Off-site doses & health effects with their frequencies (probability distribution, usually in the form of a CCDF, Complementary Cumulative Density Function):
- Off-site dispersion and dose assessment (public doses) calculations with the selected code (or codes, for comparison)
    - E.g. ARANO, VALMA (VTT); MACCS, RASCAL (NRC)
- Picking the relevant ones from all the dose results:
    - Relevant exposure modes / dose pathways (e.g. inhalation, cloudshine, groundshine; bone marrow, lungs, effective dose)
    - Chosen fractiles (typically 95 % or 99.5 %, as conditional probability after one certain release already has occurred), i.e. dose level which is exceeded only with low probability; then produce dose vs. distance curves
    - Comparison of predicted dose levels with IAEA or national criteria for initiating different protective measures
    - Distances from plant representative of possible extent of the EPZ (to find the distance up to which a limit triggering a protective measure can be exceeded with a high enough probability)
- Emergency Preparedness Zones are based on expected doses & dose level frequencies, dose limits for countermeasures, and the practical possibility to perform the countermeasures
- However, after all the technical / calculation procedures that have generated numbers about probability distribution of doses at various distances, there remains the inevitable regulatory decision about 'How safe is safe enough'. Or as a question of a hypothetical resident in the nearby area: 'How far from the NPP must I be so that my frequency of experiencing dose x is lower than y', where x is a dose limit for a protection measure (e.g. 10 mSv / 2d) and y is a regulator-set frequency limit (e.g. 10^{-7} / plant-year).
- END RESULT: Rigorously justified EPZ size, based on expectedly needed protective measures (i.e. performance-based, technology-neutral result, to use the NRC wording)

A comprehensive full-scope PSA study was performed for 5 US nuclear power plants as early as 1975 in the WASH-1400 Reactor Safety Study, or the 'Rasmussen report' (NRC 1975). The calculated data generated in WASH-1400 then served for developing emergency response plans (NUREG-0396) and then resulted in the current US regulations on EPZ sizes. WASH-1400 was also a starting point for PRA use more generally in nuclear industry.

Mancini et al. (2014) have based their study of 'current EMR (Emergency Management Requirements) approach' on the results of level 2 PSA. They study two plants: a GenII large reactor (a currently operating PWR) and a GenIII+ SMR (IRIS, of light water type). They distinguish 4 different approaches in deciding EPZ size from PSA-3 results: probability, consequences, risk (used by Mancini et al.) and cost-effectiveness (see Ch. 11 for more detailed discussion). As usually necessary when moving from one PSA level to the next one, Mancini et al. have categorized the results in order to reduce the otherwise exponential growth in the number of cases to be computed (see Figure 9 below).

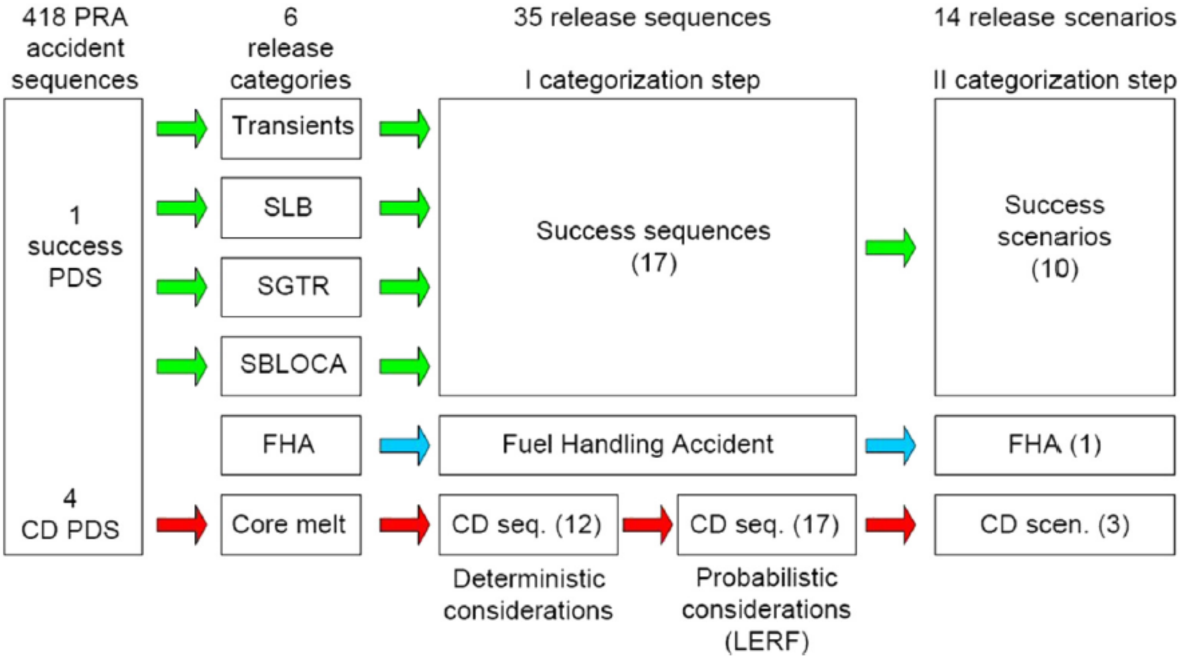


Figure 9. IRIS SMR accident sequences re-categorization by Mancini et al. (2014).

The ’NEI 2013 White Paper’ (see Ch. 5.1) on SMR EPZ includes the suggestion to use PRA methodology for SMR emergency planning, including the size of EPZ. The level of detail of the plant PRA will increase during design, construction and operation, but ‘to support the SMR EPZ size technical basis, Level 1 and Level 2 PRA should be completed’. NEI (2013) emphasizes looking at all plant operating states, which may (for SMR designs) include some ‘unconventional’ ones, like new refueling schemes, and also ‘non-core’ accidents, like fuel handling (FHA) or spent fuel pool (SFP). About offsite doses NEI (2013) states that ‘Level 3 PRA is not required, but offsite dose calculations will be necessary’.

PSA is presently used for many purposes, also in the design phase of NPPs, including SMRs (Trifanov, in ITMSR 2016), to make risk-informed decisions. It is also used during NPP operation for decisions like where money should be invested to reduce the CDF or LERF as much as possible with the given resources.

7.2 Conservative vs. BEPU, deterministic vs. probabilistic

Deterministic safety 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. For both, typical conservative assumptions e.g. for the release into the atmosphere would be its start very soon after SCRAM, and the whole release escaping the containment during a very short time interval. To be on the conservative side in PSA, e.g. the failure frequencies of systems could be taken from the higher end of estimates.

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). The change of results with respect to input should be studied by a sensitivity analysis. Uncertainties should be clearly less than the margin between the safety requirement and what is achieved by the design of the plant. Generally, with more uncertainties (from e.g. large gaps in models and data), it would be appropriate to stay on the conservative side e.g. when deciding the size of the EPZ. This could be the case for many present SMR designs.

Talabi & Fischbeck (2016) have studied the possibility of establishing SMR-specific EPZs. One of their approaches is to reduce the 'unnecessary' conservatism that was used in previous analyses. They conclude that the 'conservative approach for current plants shows that up to 30 percent of the plants in the sample exceed the EPA's Protective Action Guide limits. By improving the analysis and reducing the overly conservative assumptions, the level of confidence associated with the risk of exceedance for large plants would be improved beyond 70 percent as documented in NUREG 0396'.

NEI (2013), in the 'White Paper' on SMR EPZ, discuss uncertainty evaluations and mention steps to account for uncertainties (see NEI Ch. 4). One of the questions is 'completeness uncertainty', which means (e.g. for PRA) that some factors had to be left out, to be addressed by other approaches, because they would be too difficult to include in a rigorous manner. As one potential remedy for uncertainties, they suggest to 'show that detailed planning within the SMR EPZ provides a substantial base for expansion of response beyond the EPZ boundary'.

In their feedback questions to the NEI (2013) proposal of EPZ methodology, the NRC commented ('Question 19') on the acceptance values for the level of uncertainty in plant CDF and LERF: 'The guidance given for establishing the acceptance values is not clear'; and requested additional explanation.

Mancini et al. (2014) apply risk-informed methodology to study emergency management of a 'test case' GenII large NPP and GenIII+ SMR. They conclude that the present requirements are conservative even for large NPP, and for SMR, a reduction of the emergency area is possible without loss of safety level. They use both probabilistic and deterministic approaches to define the revised emergency management, including EPZ size, considering the area of impact and health effects on the population.

7.3 Initiating events of accidents

According to a definition by the IAEA, an initiating event of core damage (CD) is:

- An event that creates a disturbance in the plant that has the potential to lead to core damage, depending on the successful operation of required mitigating systems in the plant.

A prerequisite for PSA is a comprehensive list of all possible initiating events during plant operational modes analysed in the PSA, for example:

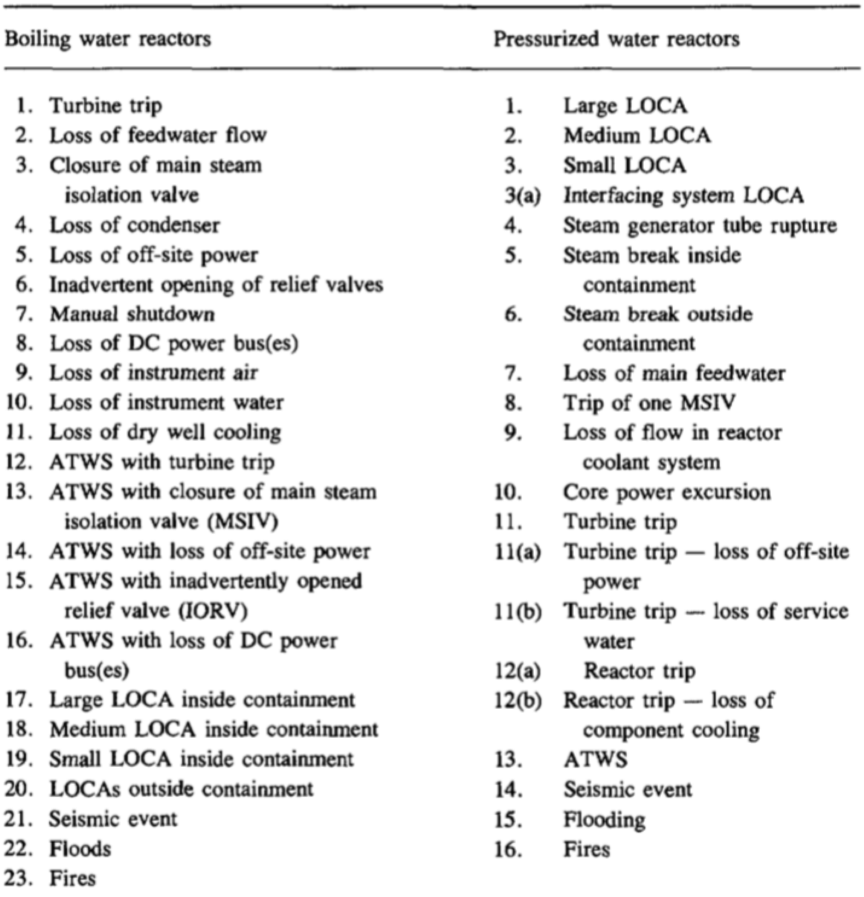
- Power operation
- Low power
- Shutdown
- Refuelling

Considering the state of cold shutdown, heating reactors might be down for long periods in the summer season. Traditionally, potential reactivity accidents during cold shutdown have been devoted special studies in safety assessments.

For emergency preparedness planning, only core damage is not enough, but the source of atmospheric radioactive releases may be also spent nuclear fuel (SNF) in the spent fuel pool (SFP) or during fuel handling operations (fuel handling accident, FHA).

The postulated initiating events (PIEs) can be divided in internal (technical, caused by the plant's own systems, or possibly human-induced) and external (outside-caused hazards, e.g. seismic or other natural) ones. The latter should include also security considerations (i.e. malevolent / hostile actions, like terrorism). See Table 9 for a sample classification of PIEs. PIE frequencies of occurrence vary. Rahn et al. (1984) mention in their Table 17.17 e.g. for PWR that loss of offsite power (LOOP), safety-related power failure, auxiliary feedwater failure and safety/relief valve failure were among the most significant potential accident sequence precursors, when looking at real plant data (operating experience) over a period of 11 years.

Table 9. Example of grouping of initiating events. Source of Table: IAEA Safety Series 050-P-4 (1992, now obsolete).



Boiling water reactors	Pressurized water reactors
1. Turbine trip	1. Large LOCA
2. Loss of feedwater flow	2. Medium LOCA
3. Closure of main steam isolation valve	3. Small LOCA
4. Loss of condenser	3(a) Interfacing system LOCA
5. Loss of off-site power	4. Steam generator tube rupture
6. Inadvertent opening of relief valves	5. Steam break inside containment
7. Manual shutdown	6. Steam break outside containment
8. Loss of DC power bus(es)	7. Loss of main feedwater
9. Loss of instrument air	8. Trip of one MSIV
10. Loss of instrument water	9. Loss of flow in reactor coolant system
11. Loss of dry well cooling	10. Core power excursion
12. ATWS with turbine trip	11. Turbine trip
13. ATWS with closure of main steam isolation valve (MSIV)	11(a) Turbine trip — loss of off-site power
14. ATWS with loss of off-site power	11(b) Turbine trip — loss of service water
15. ATWS with inadvertently opened relief valve (IORV)	12(a) Reactor trip
16. ATWS with loss of DC power bus(es)	12(b) Reactor trip — loss of component cooling
17. Large LOCA inside containment	13. ATWS
18. Medium LOCA inside containment	14. Seismic event
19. Small LOCA inside containment	15. Flooding
20. LOCAs outside containment	16. Fires
21. Seismic event	
22. Floods	
23. Fires	

IAEA SSR-2/1-Rev. 1 (Safety of Nuclear Power Plants: Design) mentions, in connection with Requirement 16, the identification of PIEs with the following words: 'The design for the nuclear power plant shall apply a systematic approach to identifying a comprehensive set of postulated initiating events (PIEs) such that all foreseeable events with the potential for serious consequences and all foreseeable events with a significant frequency of occurrence are anticipated and are considered in the design.'

Ramana & Mian (2014) observe that SMRs are said to 'offer increased safety by eliminating most accident initiators (for example, large pipes inprimary circuit), by improving decay heat removal and including more efficient passive heat removal from reactor vessel, more in-factory fabrications, transportability and site selection flexibility, smaller plant footprint and use of seismic isolators for increased seismic safety'. SMR designers estimate the resulting total CDFs (core damage frequencies) in the range of 1e-5 to 1e-8.

The NEI (2013) White paper on SMR EPZ states that FHA and SFP accidents will be considered. In the feedback questions, the NRC asked, how these accident scenarios are determined. Furthermore, NEI (2013) identifies Regulatory Guide 1.200 and American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa 2009 as the necessary guidance for demonstrating sufficient technical adequacy of the base PRA. The NRC says in feedback that the mentioned guidance does not address all relevant initiating events and operating modes, nor does it address Level 2 PRA. The NRC asks, how the lack of guidance in these additional important areas be will addressed.

NEI (2013) mention the following potential risks as difficult to address in PRA:

- security events
- collateral damage, co-location
- potential common cause effects on modules having common or shared systems
- aging effects
- organizational performance & other HFE-related risks; factors affecting operations (e.g., shift staffing, training and procedures, use of new I&C systems, and lack of operating experience)
- design faults, errors of commission
- concurrent hazards
- risks treated outside the PRA due to an endorsed standard not being available

The DiD WG of the IAEA SMR Regulators' Forum concluded in their report about internal & external hazards:

Internal and external hazards are important challenges for the DiD levels and for the independence of the levels. They can cause common mode failures that could impact the safety features involved at one DiD level and even simultaneously affect several DiD levels.

According to IAEA SSR-2/1 (Rev. 1), all foreseeable internal hazards and external hazards, including the potential for human induced events that could directly or indirectly affect the safety of the nuclear power plant shall be identified and their effects shall be evaluated. Hazards shall be considered for the determination of the postulated initiating events and of generated loadings for use in the design of relevant items important to safety for the plant.

The accident in Fukushima Daiichi NPP demonstrated that it is vital to consider the impact of common cause and common mode failures when implementing the concept of DiD, particularly from external hazards, as they can lead to a loss of several levels of DiD safety provisions or significantly reduce independent effectiveness.

Internal hazards:

An NPP should be designed with adequate physical separation (e.g., by barriers, by distance or both) to protect the safety features implemented at each of the DiD levels against all potential internal hazards (such as fires, explosions and floods).

Internationally available documentation on SMRs does not present in detail the list of postulated internal hazards, how they are considered in the design and the provisions foreseen to protect the safety functions against such hazards.

The list of internal hazards taken into account in the safety demonstration should be justified by SMR designers, considering all SMR design specifics. All potential internal hazards that may occur within the module or in areas common to multiple modules should be considered.

Particular attention should be paid in SMR design to potential common mode failures due to internal hazards (such as fires, explosions, internal flooding and load drops) and to their influence on DiD levels effectiveness and independence, taking into account the SMR design specifics (e.g., modularity, compact design and multi-units).

As stated in IAEA SSR-2/1 (Rev. 1), for multiple unit plant sites, the design shall take due account of the potential for specific hazards to give rise to impacts on several or even all units on the site simultaneously. This statement is particularly applicable to multi modules/units SMRs.

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

External hazards:

Like typical large reactors, SMRs could be threatened by their environments. Therefore, the risks of external hazards – natural or man-induced – should be taken into account in the safety assessment of SMRs, considering their specific location and environment.

Because SMRs may be located remotely or in many different environments, a detailed analysis of possible external hazards and associated risks for SMRs should be performed for each specific application.

As stated in IAEA SSR-2/1 (Rev. 1), for multiple unit plant sites, the design shall take due account of the potential for specific hazards to give rise to impacts on several or even all units on the site simultaneously. Concerning the simultaneous impacts of external hazards on several units, WENRA states that "On multi-unit sites, the plant should be considered as a whole in safety assessments and interactions between different units need to be analyzed. Hazards that may affect several units need to be identified and included in the analysis."

7.4 Multi-unit / multi-module considerations

Multiple units of an SMR plant may be, in some respects, more problematic in safety assessment than present several, but quite independent reactor units at one site. Regarding PIEs, at the same site many units may be damaged by a common cause (CCF, common cause failure). In the case of SMR plants, there may be several small units at the site, possibly in the same building and under the control of one common control room, which may introduce new HFE (human factors engineering) related risks.

In their feedback questions to NEI (2013) SMR EPZ 'White Paper', the NRC considered the frequencies (CDF, LERF) per plant-year vs. per reactor-year and asked 'how many coincident core damage events will be assumed to develop the DBA offsite dose estimates to compare against the PAGs?' Another NRC feedback inquiry was about accident source term evaluation considering credible multimodule accidents, about the NEI basis for the conclusion that 'source terms and associated dose would not be expected to be additive' for multimodule SMRs.

In 2017, the NRC considered in their ML17206A265 ('Rulemaking for Emergency Preparedness for SMR') also collocation of facilities and multi-module facilities. Collocation on the same site must be considered with SMRs of the same type, with large NPPs, and at industrial facilities (typically those served with heat and/or electricity from the SMR). The effects of collocation include the size of the EPZ. For multi-module sites, the NRC sees complex considerations, like shift staffing changes (when the number of modules increases), and the impact on modules that have common or shared systems.

The DiD WG (Defence-in-Depth working group) of the IAEA SMR Regulators' Forum (2018) gave their conclusions also about multi-module issues:

- SMR 'module' is not equivalent to the 'unit' / 'plant' of large reactors. The safety principles for the 'multi-units' issue cannot be directly applied to 'multi-modules' in SMR facilities.
- It must be demonstrated for 'multi-modules' facilities that connections, shared features and dependencies among modules are not detrimental to DiD.
- Even if the SMR concept is based on modular design with small unit power on multi modules/units sites, the SMR design should account for the potential consequences of several – or even all – units failing simultaneously due to external hazards. It may affect the methodology for EPZ assessment.

The Graded Approach WG (GA WG) described in their report the multiple-unit licensing in Canada by the regulator CNSC under the NSCA (Nuclear Safety and Control Act). A single license for all activities / facilities, with technical, age and lifecycle stage differences, on the site can be done efficiently. However, special consideration must be given to the SMR option of future installation of additional units, with possible technical differences. The applicant should consider the whole facility's ultimate total capacity over its lifetime, particularly when studying potential adverse impacts to the environment. Human factors include errors like doing something accidentally on the wrong unit.

The DiD WG listed the following to-be-investigated safety issues for multi-module facilities:

- requirements for shared systems or interconnections
- impact of multi-module configurations on the risk of propagation of an AOO, a DBA or a DEC or an internal hazard from one module to other modules
- simultaneous impact of external hazards on several modules
- confinement function
- common spent fuel pool
- human and organizational aspects
- a single control room common to several modules

NEI (2013) emphasize 'extreme site-wide situations, including potential impacts on reactor modules that have some common or shared systems, where it is difficult to foresee all potential conditions in advance'. Then the most important basic safety functions should be maintained: core cooling, electric power, containment integrity, and spent fuel pool cooling under forcing, long-lasting conditions (like permanently installed equipment being unavailable and/or the site environs being damaged with limited access).

Accounting for collocation (multiple units) has been studied at VTT by Ilkka Karanta in the PRAMEA project. Björkman & Tyrväinen (2021) have considered multi-module issues of SMRs. They find the challenges generally similar to multi-unit case. However, aspects that differentiate multi-module SMR PRA from units of LLWRs include risk metrics, plant operating states, initiating events, human dependencies and common cause failure (CCF) groups.

The European MELCOR/MACCS Users Group Meeting (EMUG) of April 2019 included a presentation by N E Bixler of Sandia SNL on application of MACCS (PSA level 3 code) to multi-unit consequence analysis. He presents BE MUPSA (Best-Estimate Multi-Unit PSA), where M units have (for each) N identical source term categories. The number of possible source term (and thus consequence) variations quickly increases with the number of units considered.

7.5 Very low frequency events

Initiating events have a vast range of possible frequencies of occurrence, from e.g. 1e-2 (events per reactor-year) of a RCP (reactor coolant pump) seal LOCA (loss of coolant accident) to 1e-7 of a catastrophic RPV (reactor pressure vessel) failure. Some very unprobable initiating events (less than a low frequency threshold) may be left out of analysis, for example a direct hit by a meteorite (the probability of which is basically a well-known figure). On the other hand, there are many unprobable external events whose probabilities / frequencies are very difficult to estimate, for example that of a successful terrorist strike.

After the Fukushima accident of March 2011, the so-called extreme events received increased attention. Tsunamis are not rare in Japan, but the height of the wave was exceptional. The simultaneous effect on several units was also something not considered thoroughly, as also the fact that the whole surrounding area was ruined by the phenomenon. In Finland, the EXWE project in SAIR2018 has studied weather phenomena that are extreme (low frequency, but big effects) by Finland standards.

NEI (2013) states that uncertainties in quantification for very low frequency accident sequences make it appropriate to evaluate the potential impact on risk (i.e. probability x consequence) of these sequences (‘cliff edge effects’).

The IAEA Regulators’ Forum (2018) concluded that ‘EPR arrangements, including EPZ&D, need to be developed based on results of hazard assessment, accounting for events of very low probability and events not considered in the design’.

About current SMR projects (SMART and HTR-PM) Ramana et al. (2013) say that ‘even if such accidents might be very unlikely, given the history of Chernobyl and Fukushima, prudence requires planning for such extreme scenarios. This may be one reason motivating some countries, such as South Korea and China, to continue with adherence to traditional requirements.’

In the UK, improvements in EPZ calculations are currently expected. In the updated approach, consideration of some lower frequency, higher consequence accidents should be required.

NRC SECY-97-020 states that ‘detailed planning within the EPZ is expected to provide a substantial base for expanding response efforts, should expansion be necessary for those low probability, high consequence events whose effects could extend beyond the EPZ’.

EPR should be made also for very low probability events, for scale encountered in the past, and for events not considered in plant design, both for accidents and malevolent security events. Enhanced safety features are expected to affect the released amounts (nuclide-specific activities), starting time (delay to onset) and duration of the release, which in turn affect the offsite doses. It must be particularly emphasized that EPR arrangements are for those cases in which the safety systems, designed for a certain range of cases, proved to be inadequate, and a release took place. So it is principally not possible to do away with EPR, however low the probabilities and consequences of the considered events might be.

After Fukushima, preparedness also for very low probability events became more important. These may also include security and military events.

7.6 New SMR-specific risks

Many current SMR designs clearly have improvements in safety features, particularly in inherent / passive ones. At the same time, there is a tendency to claim for them new, less stringent licensing requirements e.g. in emergency management, justified by the inherently improved safety. However, worries have been expressed, e.g. by the UCS (Union of Concerned Scientists) and in journal articles like Ramana et al.

(2013) that the benefits of improvements could be outweighed by the relaxation of requirements. An example would be the predictably lower offsite doses being offset by smaller EPZ and thus siting SMRs closer to densely populated urban areas, with the potential for large population doses because of the number of people exposed. Furthermore, there have been claims that in the US, the DOE would be promoting reduced EPZ and encouraging the NRC to weaken requirements to facilitate the US domestic SMR providers industry.

In addition to the possible general effect of relaxing requirements, there are some possible new risks specific to SMR, including the following:

- Integration may affect irradiation of components & ease of inspections. Integrated in the same RPV, the SGs and pressurizer will receive more radiation dose than at more remote locations. It may also be more difficult to perform inspections because of the tight packaging.
- Shared systems / buildings of several modules: There is the risk that a CCF could affect all modules at once, or that a problem at one module might spread to other ones.
- Control room: In the case of SMR plants, there may be several small units at the site, possibly in the same building and under the control of one common control room, which may introduce new HFE (human factors engineering) related risks, like doing an operation on the wrong module, or neglecting what happens in one module when focusing on another.
- Few staff present: There have been suggestions to operate some SMRs even completely unmanned, controlled from a remote location. This is the way to use some conventional small district heating plants. Continuous keeping of staff at a small plant would be a relatively big financial burden. However, it would seem risky to leave a nuclear plant to run on its own, and this is probably hard to accept by regulators.
- Industry near / at the site: Traditionally, the EPZ of large NPPs is kept free from large-scale industry which might be affected by a nuclear accident, or a conventional accident at which could endanger the NPP. SMRs may be located next to industry quite on purpose, and still the same considerations apply. On the other hand, an industrial site is usually fenced, manned and prepared for accidents with its own emergency staff.
- Transports: Some SMRs are to be factory-fueled, transported to site and then back, containing spent fuel. Transports may take place during operation of other modules. The risk of a radioactive atmospheric source term is then present, to some extent, also along the transportation route, at intermediate ports, storage points, land-based maintenance facilities etc.
- Remote areas: Even when large NPPs were traditionally located distantly from dense population centers, very remote areas may have the additional drawback of having very little emergency operating capability, should an accident happen.
- Lack of operating experience: According to the IAEA Regulators' Forum EPZ WG, 'lack of operating experience means that the safety case will have greater uncertainties that will need to be addressed by use of conservatism or additional safety and control measures'.

High level of nuclear safety and security cannot be achieved by smart design alone ('safety-by-design'). It must also include operation, however smart and safe the design is, just like new cars are relatively safe, but still the driver may cause accidents. Safety-by-design may have a particular drawback because of human nature: When cars were equipped with ESC (electronic stability control), many drivers started to increase their speeds, relying on the ESC to stabilize the vehicle under any circumstances. A similar problem was recognized by the EPZ WG as a recommended future activity: 'Examine the safety culture with respect to SMR industry. This topic arises from new designers and operators entering the industry, as well as, creating a culture from the beginning to not become complacent by safety by design'. This may not be so much of a problem in Finland, with high education level and existing good tradition of nuclear energy, but may be so in emerging nuclear energy countries, particularly in developing countries (which have some specific reasons, like financing or grid capacity, to consider SMR).

8. Full-scope PSA: Source terms

For level 3 PSA, included in a full-scope PSA study and strongly recommended in this report, the strating point is the atmospheric source, i.e. radionuclides escaping from the containment to the atmosphere. Compared with othe PSA3 input data, it is usually regarded as the most uncertain one. Site-specific weather data can be more accurate, and the processes of atmospheric dispersion & dose assessment can usually be predicted better than those of severe accidents & containment phenomena.

The source term should include the following data for each sequence of accident & containment by-pass:

- The frequency of the source term occurring
- Released activities (Bq) of important nuclides (100 is typically well sufficient)
- Chemical form of the nuclides, particularly iodine (gas, alkali salt, organic bound etc.)
- Effective release height (or heights as function of time). Ranging from a hole in containment wall to a release through ventilation stack. The so-called cloud rise may increase effective release height because of heat content or initial upward momentum. A high release is usually less dangerous from the point of view of a nearby individual person.
- Timing of release, i.e. duration and release rates as function of time. Typically the activities refer to the time of SCRAMming the reactor, and the subsequent chain decay is calculated separately. A long-lasting release usually makes the radioactive plume more dilute, compared with all the stuff released as an instantaneous puff.

Some nuclides with particular radiological importance are exemplified with the following list, with boiling point included:

Iodine	I	+184	131, 132, 133, 134, 135
Cesium	Cs	+671	134, 136, 137
Krypton	Kr	-153	85, 85m, 87, 88
Ruthenium	Ru	+4150	103, 105, 106
Tellurium	Te	+988	127, 127m, 129, 129m, 131m, 132

Released activity of each nuclide depends on reactor inventory at shutdown and the actual release fraction, which is affected by the volatility / boiling point. Hummel et al. (2020) have used models of Arrhenius form to express release rates as a function of temperature. After shutdown the activities follow from chain decay with various half-lives and daughter nuclides. After dispersion, the relative importance of each nuclide with regard to dose consequences finally follows from dose conversion coefficients in the considered dose pathways.

8.1 SMR core radioactive inventories

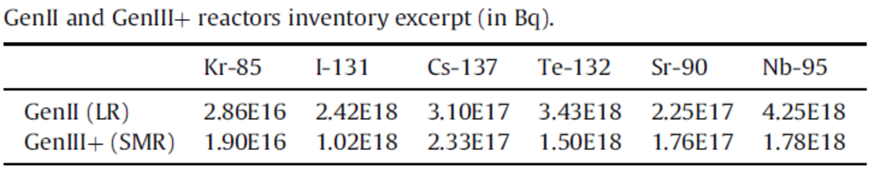
An atmospheric radioactive release contains an accident-specific fraction (basically different for each element, or element group) of the core radioactive inventory. Therefore, the starting point of the PSA1 to PSA3 study, or more detailed deterministic calculations by severe accident assessment codes, should be the nuclide-specific activity inventory. It will change when moving from beginning-of-cycle (BoC) to end-of-cycle (EoC), and is usually different for the first fuel cycles after commissioning. For the purposes of off-site dose assessment considered here, it could be the time-average inventory, or a ‘worst case’ one from EoC (high burn-ups). In any case, the inventory of the SMR to be investigated should be calculated, if not acquired directly from its designer.

The reactor core radioactive inventory can be calculated e.g. using the Serpent Monte Carlo neutronics code. In the past similar tasks were usually done with the Origen code. Serpent has been used in 2016 for

NuScale calculations, where a test case mock-up SMR core in a steady state at full power with single phase flow was modelled. Neutronics was solved with Serpent 2 code and thermal-hydraulics with COSY (Component/System-scale) thermal-hydraulics (TH) tool. Both of the codes are developed at VTT. More information about the 2016 work can be found in VTT-R-05548-16 [Hillberg et al. 2016]. The inventory results can be briefly described as follows:

- NuScale resembles usual LWR reactors > no big differences expected in relative nuclide composition
- Nuclide-specific activities should be roughly scalable by power level.
- Serpent results for 1380 nuclides & 46 burn-up time points (to 50 MWd/kgU)
- Some approximate comparison with Olkiluoto-3:
    - NuScale electric output 45 MWe (= 1/36 of Olkiluoto-3)
    - NuScale 50 MWd Kr-85 (11 a) 4.1e15 Bq (1/13 of OL-3 average)
    - NuScale 50 MWd I-131 (8 d) 1.7e17 Bq (1/25 of OL-3 average)

Mancini et al. (2014) have calculated EPZ results for both SMR and large reactor, but their inventories of the two facilities differ from each other only by a small factor (table below):



GenII and GenIII+ reactors inventory excerpt (in Bq).

	Kr-85	I-131	Cs-137	Te-132	Sr-90	Nb-95
GenII (LR)	2.86E16	2.42E18	3.10E17	3.43E18	2.25E17	4.25E18
GenIII+ (SMR)	1.90E16	1.02E18	2.33E17	1.50E18	1.76E17	1.78E18

8.2 Codes for severe accident assessment

Even when the ‘big picture’ of accident sequencies and their frequencies (and the resulting atmospheric releases and their frequencies) is studied by probabilistic methods (full-scope PSA with levels 1-2-3), it is necessary to use deterministic codes, calculating all the relevant physical phenomena, to generate detailed information on how the radioactive nuclides from leaking/molten fuel are transported within the primary circuit, then inside the reactor containment and finally (possibly) in the atmosphere. Severe accident (SA) assessment codes, like MELCOR, can usually predict the processes taking place before the atmospheric release. For offsite doses, specialized atmospheric dispersion and dose assessment codes are used.

Examples of well-known SA codes are MELCOR, ASTEC and MAAP. With the current ‘SMR wave’ internationally, questions have been raised about the suitability of current SA codes (usually best tailored to current large LWRs) for SMRs. The main problems could be the passive systems used in many designs, and also evidently the more ‘exotic’ types of SMRs than LWR.

NEI (2013) on SMR EPZ names ‘the fully integrated, engineering level, severe accident analysis computer code, MELCOR, industry’s MAAP5 code, and the NRC’s State-of-the-Art Reactor Consequence Analysis (SOARCA) study as examples of usage of advanced tools and models’.

In their feedback questions to NEI, the NRC wanted to know more about the suggested SMR use of Methods for Estimation of Leakages and Consequences of Releases (MELCOR), Modular Accident Analysis Program Version 5 (MAAP5), and the State-of-the-Art Reactor Consequence Analysis (SOARCA): ‘These tools and models are designed for large light water reactors (LWRs). Although SMRs are conceptually similar to LWRs, there is little operational or experimental data. Please provide an explanation of how these tools and models can be extrapolated for SMR analysis to inform plume exposure EPZ size.’

Possibility of a severe accident (one with core melting at least partially) should always be considered (however improbable it may be). This is justified when considering the European requirements given in Euratom Safety Directives (2013/59, Basic safety standards), and (2009/71, Community framework for nuclear safety), amended by 2014/87.

Even with inherent safety features, severe accidents cannot be neglected. In Finland, no new-build nuclear power plant is acceptable without a feasible strategy for managing severe accidents (STUK YVL 2.2, old, and B.3 Deterministic Safety Analyses & B.6 Containment, new). Differences of SMRs from large power reactors include integrated RPVs, lower power levels and smaller reactor core radioactive inventories. Melt coolability and containment heat removal are still important issues.

Severe accidents are unlikely but have to be taken into account in the design of all modern power reactors. Severe accident management (SAM) and mitigation measures are being required of the licensees by the national regulator. In Finland, STUK requires that severe accidents shall be considered in the planning of new-build reactors. A short description is included here on severe accident phenomena in general. The reader can refer to VTT-R-05548-16 [Hillberg et al. 2016] for more information. At VTT, severe accident simulations are mainly conducted by using the integral codes ASTEC and MELCOR.

In the first RCM (research coordination meeting) of IAEA CRP I31029 in May 2018, one of the focus areas identified for SMR EPZ determination was the identification of the events, severe accidents and progression (timing and evolution) leading to a source term that is released from the facility to the atmosphere. A key piece of information is what barriers expected to be in place in the case of credible accidents.

The phenomena and management of a postulated severe accident include:

- Cooling of the molten core
- Formation of hydrogen and the related combustion risk
- Release and transport of radioactive fission products
- Direct heating of the containment
- Energetic fuel-coolant interactions (steam explosions)
- Containment pressurization
- Long-term decay heat removal
- Re-criticality

Active and passive safety systems, structures or components can be distinguished from each other by determining whether there exists any reliance on external mechanical or electrical power, signals or forces in their functioning:

- With no such external connections, natural laws, properties of materials and internally stored energy are used instead (passive safety).
- Some potential causes of failure of active systems, such as lack of human action or power failure, do not exist in passive safety (Safety related terms for advanced nuclear plants, 1991).

The defence-in-depth principle is the basis of the safety design of SMRs and also the foundation of the Finnish regulatory guidelines of nuclear safety. The passive decay heat removal safety systems, featured in many SMRs, are taken into account in Finnish regulations by giving them a reduced failure criterion (N+1) compared to (N+2) for active systems.

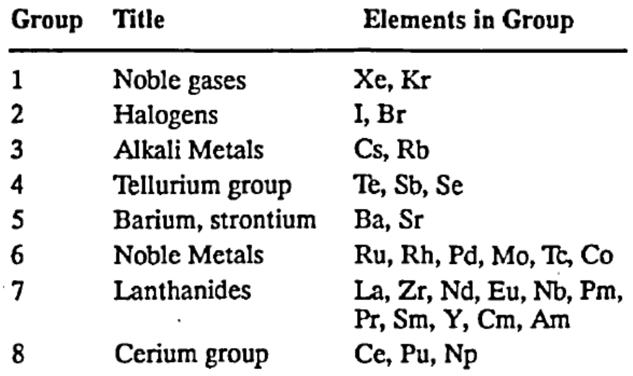
8.3 Atmospheric source terms from large LWR

There is a considerable amount of accumulated knowledge of atmospheric radiaoactive source terms caused by accidents at large reactors, mostly ‘traditional’ LWR (of PWR or BWR type). Most of the data was generated by theoretical studies, but a few cases exist where an accidental release actually happened and some measurements are available. However, considering all input data of PSA level 3, the source term is probably the one with most uncertainty, despite all the efforts spent to determine it for various accidents. At VTT, such studies have been done e.g. with the MELCOR code.

The NRC has published several guides on the source term, including the following:

- NUREG-1465 (1995): Accident Source Terms for Light-Water Nuclear Power Plants; ML041040063.pdf
    - In-containment source terms, available for leakage from the reactor to the environment.
    - Accident Sequences Reviewed
    - Onset of Fission Product Release; Duration of Release Phases
    - Fission Product Composition and Magnitude; Chemical Form (cf. Table 10)
    - Proposed Accident Source Terms

Table 10. Radionuclide grouping used in NUREG-1465 (mainly according to chemical characteristics).



Group	Title	Elements in Group
1	Noble gases	Xe, Kr
2	Halogens	I, Br
3	Alkali Metals	Cs, Rb
4	Tellurium group	Te, Sb, Se
5	Barium, strontium	Ba, Sr
6	Noble Metals	Ru, Rh, Pd, Mo, Tc, Co
7	Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
8	Cerium group	Ce, Pu, Np

- RG-1.183 (2000): Alternative radiological source terms for evaluating DBAs at nuclear power reactors; ML003716792.pdf
    - Fission Product Inventory; Release Fractions
    - Timing of Release Phases
    - Radionuclide Composition; Chemical Form
    - Fuel Damage in Non-LOCA DBAs
    - Offsite Dose Consequences; Acceptance Criteria
    - Meteorology Assumptions
    - LOCA, FHA, BWR rod drop, BWR & PWR MSLB, SGTR, PWR locked rotor, PWR rod ejection

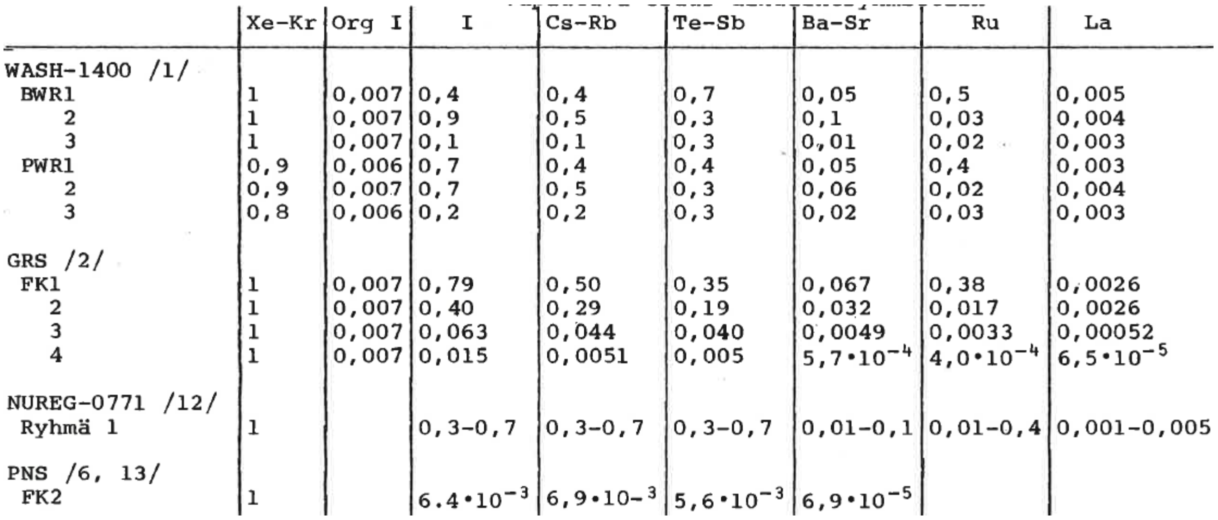
- Technical basis for revised RG-1.183: Fission product fuel-to-cladding gap inventory (2009); ML090360256.pdf
    - Fission product gap inventories for DBA

The so-called total decontamination factor (DF) means how much of a radionuclide escaping from the fuel will finally reach the atmosphere, if it escapes the containment. Currently in the US (Regulatory Guide 1.183) correlations for aerosol natural deposition may be used. These are not the same for large LWR and SMR. Probably the DF would be better for SMR because of the smaller ratio of volume to surface area.

Experiments should be performed to study particularly the processes of diffusiophoresis & thermophoresis and hygroscopic effects.

Some representative release fractions for a large LWR, partly based on the WASH-1400 report (NRC 1975) and NUREG-0771 (Pasedag et al. 1981) are given in Table 11 below:

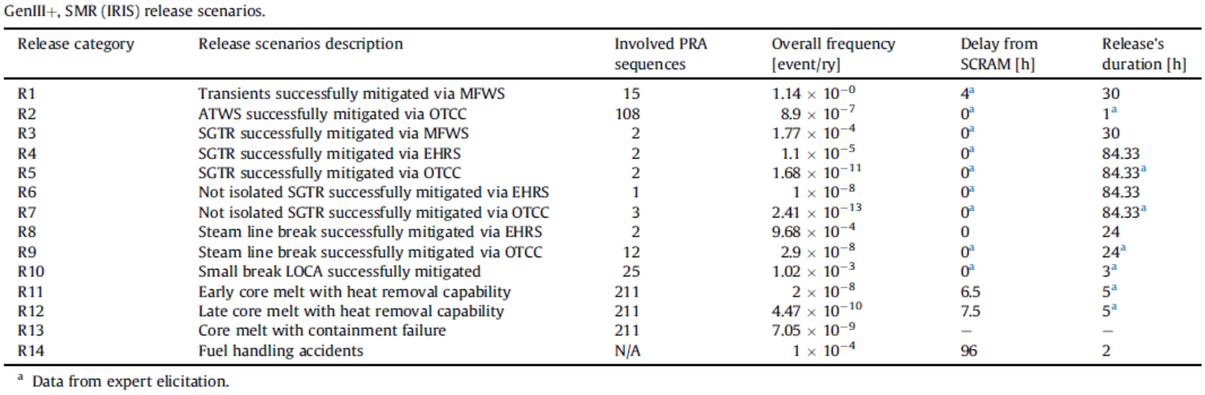
Table 11. Release fractions (of core radioactive inventory) used in the WASH-1400 report.



	Xe-Kr	Org I	I	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
WASH-1400 /1/								
BWR1	1	0,007	0,4	0,4	0,7	0,05	0,5	0,005
2	1	0,007	0,9	0,5	0,3	0,1	0,03	0,004
3	1	0,007	0,1	0,1	0,3	0,01	0,02	0,003
PWR1	0,9	0,006	0,7	0,4	0,4	0,05	0,4	0,003
2	0,9	0,007	0,7	0,5	0,3	0,06	0,02	0,004
3	0,8	0,006	0,2	0,2	0,3	0,02	0,03	0,003
GRS /2/								
FK1	1	0,007	0,79	0,50	0,35	0,067	0,38	0,0026
2	1	0,007	0,40	0,29	0,19	0,032	0,017	0,0026
3	1	0,007	0,063	0,044	0,040	0,0049	0,0033	0,00052
4	1	0,007	0,015	0,0051	0,005	$5,7 \cdot 10^{-4}$	$4,0 \cdot 10^{-4}$	$6,5 \cdot 10^{-5}$
NUREG-0771 /12/								
Ryhmä 1	1		0,3-0,7	0,3-0,7	0,3-0,7	0,01-0,1	0,01-0,4	0,001-0,005
PNS /6, 13/								
FK2	1		$6.4 \cdot 10^{-3}$	$6,9 \cdot 10^{-3}$	$5,6 \cdot 10^{-3}$	$6,9 \cdot 10^{-5}$		

Mancini et al. (2014) studied offsite doses for EPZ size determination for both SMR and large NPP. The release fractions (of core inventory) for release categories R1-R12 defined in their study (see Table 12) for the large NPP were as shown in Table 13 below.

Table 12. GenII large reactor accident sequences used by Mancini et al. (2014).

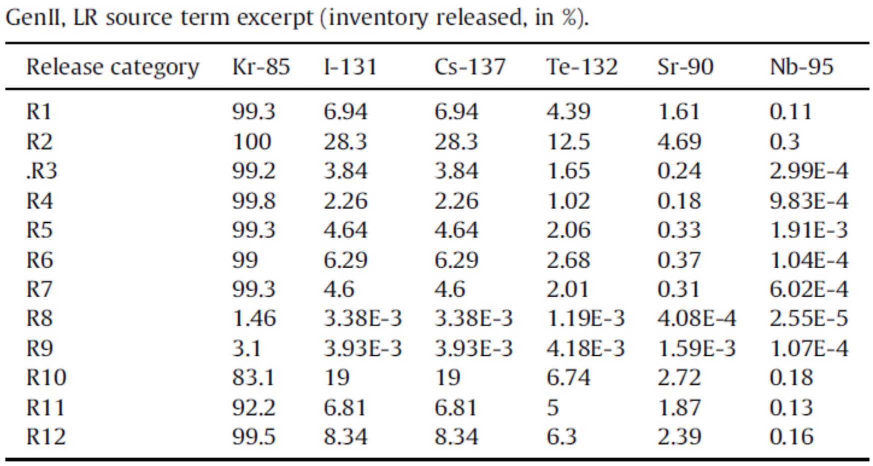


GenIII+, SMR (IRIS) release scenarios.

Release category	Release scenarios description	Involved PRA sequences	Overall frequency [event/ry]	Delay from SCRAM [h]	Release's duration [h]
R1	Transients successfully mitigated via MFWS	15	1.14×10^{-0}	4[a]	30
R2	ATWS successfully mitigated via OTCC	108	8.9×10^{-7}	0[a]	1[a]
R3	SGTR successfully mitigated via MFWS	2	1.77×10^{-4}	0[a]	30
R4	SGTR successfully mitigated via EHRS	2	1.1×10^{-5}	0[a]	84.33
R5	SGTR successfully mitigated via OTCC	2	1.68×10^{-11}	0[a]	84.33[a]
R6	Not isolated SGTR successfully mitigated via EHRS	1	1×10^{-8}	0[a]	84.33
R7	Not isolated SGTR successfully mitigated via OTCC	3	2.41×10^{-13}	0[a]	84.33[a]
R8	Steam line break successfully mitigated via EHRS	2	9.68×10^{-4}	0	24
R9	Steam line break successfully mitigated via OTCC	12	2.9×10^{-8}	0[a]	24[a]
R10	Small break LOCA successfully mitigated	25	1.02×10^{-3}	0[a]	3[a]
R11	Early core melt with heat removal capability	211	2×10^{-8}	6.5	5[a]
R12	Late core melt with heat removal capability	211	4.47×10^{-10}	7.5	5[a]
R13	Core melt with containment failure	211	7.05×10^{-9}	–	–
R14	Fuel handling accidents	N/A	1×10^{-4}	96	2

[a] Data from expert elicitation.

Table 13. GenII large reactor atmospheric source terms used by Mancini et al. (2014).



GenII, LR source term excerpt (inventory released, in %).

Release category	Kr-85	I-131	Cs-137	Te-132	Sr-90	Nb-95
R1	99.3	6.94	6.94	4.39	1.61	0.11
R2	100	28.3	28.3	12.5	4.69	0.3
.R3	99.2	3.84	3.84	1.65	0.24	2.99E-4
R4	99.8	2.26	2.26	1.02	0.18	9.83E-4
R5	99.3	4.64	4.64	2.06	0.33	1.91E-3
R6	99	6.29	6.29	2.68	0.37	1.04E-4
R7	99.3	4.6	4.6	2.01	0.31	6.02E-4
R8	1.46	3.38E-3	3.38E-3	1.19E-3	4.08E-4	2.55E-5
R9	3.1	3.93E-3	3.93E-3	4.18E-3	1.59E-3	1.07E-4
R10	83.1	19	19	6.74	2.72	0.18
R11	92.2	6.81	6.81	5	1.87	0.13
R12	99.5	8.34	8.34	6.3	2.39	0.16

8.4 SMR atmospheric source terms

The atmospheric source term could be defined by a variety of methods, ranging from expert judgement (based on previous knowledge of reactors with somewhat similar technology) at the ‘vague’ end, to ‘mechanistic source term’ (MST) modelling starting from physical principles of all the phenomena / processes present, at the sophisticated & detailed end of the spectrum of choices. Compared with large NPPs, the source terms from SMR accidents may have some general differences:

- Main difference: Smaller reactor core radioactive inventory, generally simply because of the lower level of thermal power. As a first approximation, the inventory is proportional to power.
- Nuclide content and chemical form may be different, particularly when considering other than LWR kind of SMRs.
- Due to the improved safety systems, delay from accident start / reactor SCRAM to start of atmospheric release may be longer, meaning more time for decay.
- More scrubbing within containment (more surface per volume, large water volumes etc.), meaning higher DF (better decontamination factor)
- The release might be divided along a longer interval of time. Longer release duration makes the radioactive cloud more dilute, but may, on the other hand, require some protective measures over larger areas, as the wind direction changes. Preparedness should exist for response in all directions. The release rate will be a function of time.
- Many SMR designs locate the reactor underground, or at least the size of the building is smaller than for large NPPs. Possibly the initial heat content is smaller. This all will make the effective release height smaller, meaning a more concentrated radioactive cloud in the nearby areas.

NRC SECY-93-092, refered to also in ‘Rulemaking for Emergency Preparedness for SMR and Other New Technologies’ (NRC 2017), defines MST as follows:

‘A mechanistic source term is the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.’

The successive (and decreasing) amounts / activities to be modelled, starting from fuel, are:

- Activity inventory in the fuel
- Release from fuel
- Release to containment
- Release to the environment (atmosphere)

In the case of SMR smaller containment volume, the larger ratio of surface area to volume may lead to increased aerosol deposition having positive effect on potential source term reduction. Same kind of effect may be caused by the relatively large water volumes.

After all, the potential dangers of nuclear power are caused by the radioactive fission products contained in the fuel. Ingersoll (2009) writes about the reduced source term (smaller core inventory): ‘Fundamentally, reactor safety comes down to ensuring that a dangerous amount of radiation is not released to operations personnel or the general public. The total quantity of radionuclides in a reactor core that is available for potential dispersion is referred to as the reactor ‘‘source term’’ and is roughly proportional to the power level - twice the power level yields twice the source term. The benefits of a lower source term can be manifested in a number of ways, including reduced shielding, reduced site boundary, or reduced emergency planning zone (EPZ). The latter benefit also adds flexibility for co-locating the plant for process heat applications.’

Mancini et al. have used source terms for the IRIS SMR in the EU-developed RODOS dispersion and dose assessment system: ‘RODOS takes into account the main six families of inventory isotopes (Xe, I, Cs, Te, Ba, Ru-La-Ce). Since the IRIS total inventory is not available in RODOS database, the source term is given directly as an input, in terms of activity (in Bq) for each isotope, up to 24 different isotopes (the maximum number of input isotopes in RODOS). These values are obtained from deterministic calculations of each release scenario for IRIS, performed by means of RADTRAD code runs. RADTRAD is a simplified code with respect to the modelling in RODOS, but it is able to perform both internal and external NPP release analysis. The source terms have been extracted from the RADTRAD outputs, considering all the release pathways from the core to the NSSS boundary. RADTRAD requires the definition of pathways among the compartments, for the IRIS model it can be envisioned 13 pathways.’

For all possible source terms to be used in the study of EPZ determination, the definition should ideally consist of nuclide-specific released activities, temporal distribution of release rates (Bq/s) and their initial (before any considerable atmospheric transport) distribution along the vertical (height) axis at each time point.

The Fukushima disaster made the EPR community once again pay attention to long-lasting releases. Usually the early phase of emergency should be over as soon as the radioactive cloud has passed, but in Fukushima the releases lasted for days at varying release rates.

Estimates of the atmospheric release source term, from rough estimate to more advanced, but still very uncertain ones, can be obtained for an EPZ determination study at least in one of the following ways:

- Direct information from designer
- Expert judgement of release fractions, starting from core radioactive inventory
- Deterministic computational assessment using an integral code (e.g. MELCOR)
- Probabilistic assessment (PSA level 2) to have information on possible source terms: their magnitudes (nuclide-wise activities in Bq) and also frequencies

9. Full-scope PSA: Dispersion of radionuclides

9.1 Dispersion in the atmosphere, groundwater and marine environment

Radioactive releases to the environment from the containment of the nuclear reactor can spread by atmospheric dispersion, through groundwater, or in the sea if considering a marine-based SMR. Practically only atmospheric releases may necessitate offsite emergency response. Some protective actions may be needed also in the other dispersion paths.

The main processes to be considered to model the atmospheric dispersion of radionuclides are:

- Source of emission
    - Point source, or a larger area or volume
    - Instantaneous vs. continuous release (puff vs. plume)
    - Effective release height ('effective stack height' in Figure 10)
    - Stack-tip downwash (high-rise structures causing downwards transfer of activity concentrations)
    - Building wake effects
- Physical and chemical properties of the radionuclides
    - Radioactive chain decay, with generation of daughter nuclides
    - Formation of aerosol particles of different sizes
    - Chemical reactions
    - Wet ('rainout' and 'washout' in Figure 10) & dry deposition to ground surface
- Meteorological factors
    - Wind speed and direction (actually a 3D time-dependent field)
    - Atmospheric turbulence (roughly described by Pasquill stability class)
        - Temperature gradient in the vertical direction
    - Humidity and precipitation (rain, snow; washout coefficient)
- Characteristics of the terrain
    - Roughness of the terrain
    - Thermal properties of the ground surface
    - Buildings & other obstacles
- Turbulent dispersion ('turbulent diffusion'); see 'eddies' in Figure 10
    - Mixing (dilution) caused by turbulent flows in the atmosphere
- In the turbulent flow, the macroscopic flow field has eddies (swirls, vortices)
- Atmospheric turbulence is caused:
    - Mechanically (friction between wind and surface)
    - Thermally (buoyant forces, with heating from the sun)
- Atmospheric turbulence increases with:
    - Wind speed
    - Terrain roughness
    - Decreasing stability (most turbulence with A-unstable)
- Stability classes (Pasquill)
    - Stable (E, F)
    - Neutral (D)
    - Unstable (A, B, C)
    - Reaction of an imaginary air parcel displaced vertically depends on atmospheric stability. In unstable conditions, the parcel tends to continue the displaced movement, causing turbulent mixed layer.
- Mixing height (from ground to inversion layer height)
    - Higher mixed layer in unstable conditions

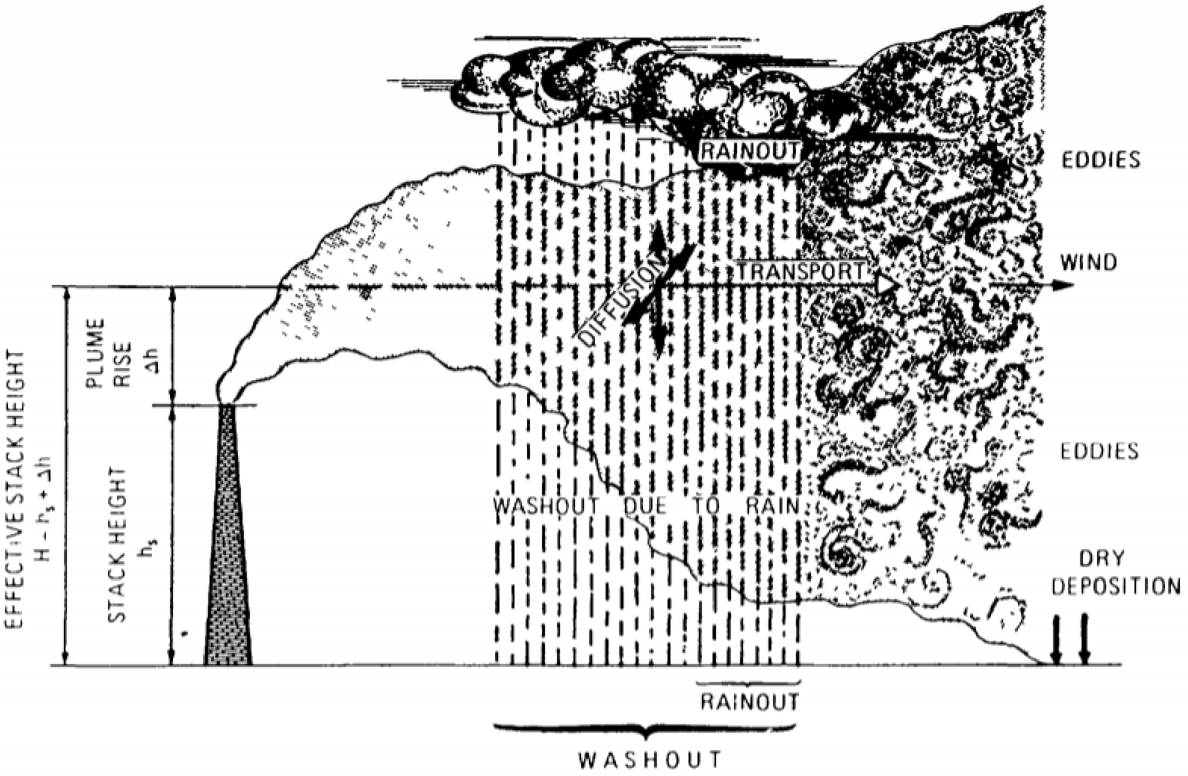


Figure 10. The main physical processes in atmospheric dispersion. Source of Figure: IAEA-TECDOC-379 (1986).

9.2 Importance of near-field dispersion effects for SMR

Many SMR designs locate the reactor underground, or at least the size of the building is smaller than for large NPPs. Possibly the initial heat content is smaller. These factors will make the effective release height smaller, meaning a more concentrated radioactive cloud in the nearby areas. Furthermore, the possible offsite adverse effects are inherently located more pronouncedly in the nearby areas only, because of the small source. Potential siting near population centers makes it very important to study those near-field effects in great detail.

Effects / phenomena important very near the source include:

- Building wake: Turbulent eddies around buildings provide more initial spread. In simple Gaussian models, like ARANO, this is accounted for by using the so-called virtual source principle.
- Stack-tip downwash (high-rise structures causing downwards transfer of activity concentrations)
- Cloud rise (plume rise): Initial upward momentum and heat content make the release rise higher than stack height right at the starting point.
- Near-field dispersion parameters: Weather mast measurements may be more reliable near the source than NWP (numerical weather prediction) model, but more masts than one single would make the data even more reliable and complete.
- Very narrow plume: It takes some time for the release to spread in the lateral and vertical directions, and when e.g. using the Gaussian model, choice of appropriate parameters is very important.
- Urban / industrial terrain: Terrain roughness may decrease wind speeds locally and induce turbulence. Furthermore, spread directions may favor street ‘canyons’ etc. In Finland, the FMI (Finnish Meteorological Institute) has developed dispersion models specifically for urban areas. One of the models is ‘CFD grade’ and requires high-performance computing capacity.

Canada (cf. Chapter 5.3) is a good example of SMR-friendly licensing process and also provides many opportunities for SMR use in remote areas, mining sector etc. The size of the EPZ is flexible and use of full-scope PSA is encouraged (see pages 13-26 'Canada' in the report of the IAEA Regulators' Forum EPZ WG).

CNL (Canadian Nuclear Laboratories) is developing a fine resolution dispersion model with any user-defined spatial and time grids for concentration predictions in research use. The spatial grid could be e.g. 5 m x 5 m. Their reasoning is that 'Assumptions of Gaussian plume model are less valid at close distances of proposed SMR EPZ, which will require more dynamic, time dependant, fine resolution dispersion model, e.g. near-field puff model.' The CNL have 'on-going work on hybrid particle/multi-puff models designed for near-field and wide particle size distribution'. (Note: The VTT-developed VALMA model can be characterized as hybrid particle/multi-puff, but mainly targeted for medium & long distance predictions.)

According to N E Bixler (17 January 2019), there is a planned MACCS code development to address near-field atmospheric transport phenomena, which is relatively more important for small-sized SMR than for large reactors.

9.3 Choice of weather data for atmospheric dispersion

If doing deterministic worst-case offsite calculations, it is customary to choose a few 'worst case weather situations', like extremely stable atmospheric conditions and/or heavily rainy weather (causing wet deposition). But for level 3 PSA studies, site-specific long-term (5 or 10 years) weather data is needed to account for the probability distributions of the various weather parameters. In VTT-made codes, ARANO categorizes the weather measurements in 1008 discrete classes, whereas in VALMA, each condition of dispersion is different because of SILAM-based weather or mast-measured but changing weather.

Mancini et al. (2014) use the EU-developed RODOS code for offsite dose predictions. The meteorological input data are: wind direction and speed, rain intensity, stability class and cloud cover. They have identified for their analysis two worst weather conditions:

- 'a moderately intense, continuous rainfall' - wet deposition causes groundshine and ingestion doses via the foodchain
- 'a prolonged atmospheric stability' with no rain - slow transport, little spread, little deposition scavenging before reaching also farther distances

Weather mast measurements may be more reliable to acquire the dispersion parameters near the source than NWP (numerical weather prediction) model, but more masts than one single would make the data even more reliable and complete. In Finland, Loviisa NPP has two weather measurement points, one at the NPP site and another out on an island in the sea for better capturing of land/sea special effects.

Selecting and acquiring representative weather data is a crucial step if a site-specific determination of appropriate EPZ size is performed:

- Existing site weather mast data typically has:
  - wind speed & direction, rain, stability
  - several years at 1 h intervals
- The Finnish meteorological institute (FMI) can provide NWP-based (numerical weather prediction) data, containing e.g. 3D wind fields as function of time, for any geographical location, through their access to WMO and ECMWF data. (Note however: The codes ARANO, MACCS and RASCAL cannot use complete 3D weather data, but only a subset thereof, whereas VALMA can utilize the 3D data.)
- Other sources of real actual weather data exist, but are usually country-specific.

Talabi & Fischbeck (2016) mention ‘improved and updated meteorological data since 1978 assessment’ as a potential basis for their ‘scalable SMR-specific EPZ’. This should probably be interpreted as a way to move from conservative assumptions to best-estimate prediction.

9.4 Site-specific considerations in dispersion & doses

There are many ways in which the site-specific environmental conditions at a nuclear plant site may affect the full-scope PSA study:

- External hazards, either natural (e.g. earthquakes, floods, sandstorms, even fauna) or human-induced (e.g. airplane crash, terrorist attack, act of war) may cause PIEs at level 1 PSA. Some of these may be very difficult to consider by PSA methods.
- Dispersion of radioactive nuclides is affected by the local weather conditions: prevailing winds, atmospheric stability, occurrence of rain etc.
- Depending on soil characteristics and rain, deposition may be resuspended into the air by e.g. high winds or sand storms, or migrate deeper into the soil, which is generally the case in Finland.
- Doses are affected by local habits (people staying more indoors or outdoors, airtightness and shielding factors of the buildings, or consumption of locally produced foodstuff).
- Effectiveness of protective measures depends on population centers, traffic connections, existence of hospitals / elderly homes / schools etc. and even communications efficiency and people’s education level.

Consideration of site-specific conditions should be included if a possible site for the plant is known. However, SMRs may be different from traditional large NPPs in this sense because they may be pre-fabricated, then sold to users and transported to the site. Then a certain SMR design could be ‘type-certified’ or ‘type-accepted’ and the siting considerations at the deployment phase could include only those things that are actually affected by the siting choice (if any such exist). Note: The NRC may grant an ESP (early site permit) to deem a site basically suitable for certain NPP types.

10. Full-scope PSA: Offsite doses

10.1 Exposure modes (dose pathways)

In dose assessment, the exposure modes (dose pathways) are categorized as internal and external. In the external ones (cloudshine, groundshine) gamma emitting radionuclides pose most danger, but also skin beta is usually taken into account. The internal pathways are typically inhalation and ingestion, and the dose is caused over the years by radionuclides that remain in the body (‘dose commitment’). Internally, alpha, beta and gamma radiation are all dangerous. Another distinction can be made by the time duration of exposure. Cloudshine and inhalation are effective only during passage of the radioactive cloud, whereas groundshine and ingestion of contaminated foodstuff may persist for tens of years (e.g. half-life of Cs-137: 30 years). The doses may be calculated as effective dose (weighted average of organ-specific doses) or doses to certain organs of most interest, like lungs, bone marrow or unborn fetus.

Typical probabilistic (based on a large number of calculation cases) offsite dose results can be expressed (for each combination of exposure pathway, distance and time point) as a CCDF curve (complementary cumulative density function), giving the probability of exceeding any dose value which appeared in the set of results (i.e. among all calculation cases). Usually a vast amount of numerical dose results will be generated: different locations (or distances), time points, internal & external exposure pathways, and all this for the various combinations of source term & weather conditions. Picking of relevant dose results must be done:

- Relevant dose pathways (inhalation, cloudshine, groundshine; bone marrow, lungs)
- Relevant exposure time periods (e.g. those appearing in the criteria for various radiological countermeasures)
- Chosen fractiles (95 % or 99.5 %), i.e. dose level which is exceeded with low probability, when the cases are those calculated with all available atmospheric source terms and weather situations.
- Distances from plant representative of possible (to be determined) extent of the EPZ

A possible first and interesting comparison would be to compare the SMR offsite doses with those from a generic large power reactor (size of e.g. Olkiluoto-3).

10.2 SMR and other specific dose considerations

In many dose pathways (exposure modes), there are some considerations (to perform an acceptably accurate assessment) originating specifically from the potential siting and small size of SMR plants:

- Cloudshine: In many codes, the so-called ‘semi-infinite uniform’ approximation of the radioactive cloud is used. Then the radiation source is a uniformly distributed concentration in a half-space (or practically, a half-sphere). With SMR plants, we are interested in the close surroundings, and it is very important to consider the cloudshine source in all of its physical reality: in some cases, a long, narrow plume extending as such relatively far (even several kilometers) from the source (cf. the LINCON code used to generate cloud gamma dose conversion coefficients for ARANO).
- Groundshine: Near population center and in industrial settings, deposition will not accumulate only on ground surface and vegetation, but to a relatively large extent also on the surfaces of buildings. They will become an additional source of radiation. Their decontamination is usually laborious.
- Inhalation: In urban environment, with a large population in a large number of buildings, the inhalation dose caused by inhaled radionuclides indoors is a particularly important consideration. Air-tightness and ventilation systems of the buildings determine how much activity will be inhaled indoors. Depending on the systems, it may be possible to reduce the internal dose commitment.
- Ingestion: Traditionally, the conservative assumption of all consumed foodstuff being produced locally, very near the target population, has been used. However, this is seldom true in today’s world, where foodstuff are routinely transported over great distances before being offered to

consumers. It is even less true for SMR surrounding environment which may be urban or industrial, and in any case the area affected by significant deposition is expected to be smaller.

Chemical forms of nuclides may be very different for other than LWR type of SMRs, but even for this ‘traditional’ kind particularly iodine may be affected by the pH and temperature of the water volumes. Possibilities include gaseous I_2, soluble salts like CsI, and organic compounds like CH_3I. The dose assessment phase has differences depending on the chemical form.

10.3 Offsite dose assessment codes

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 at VTT:

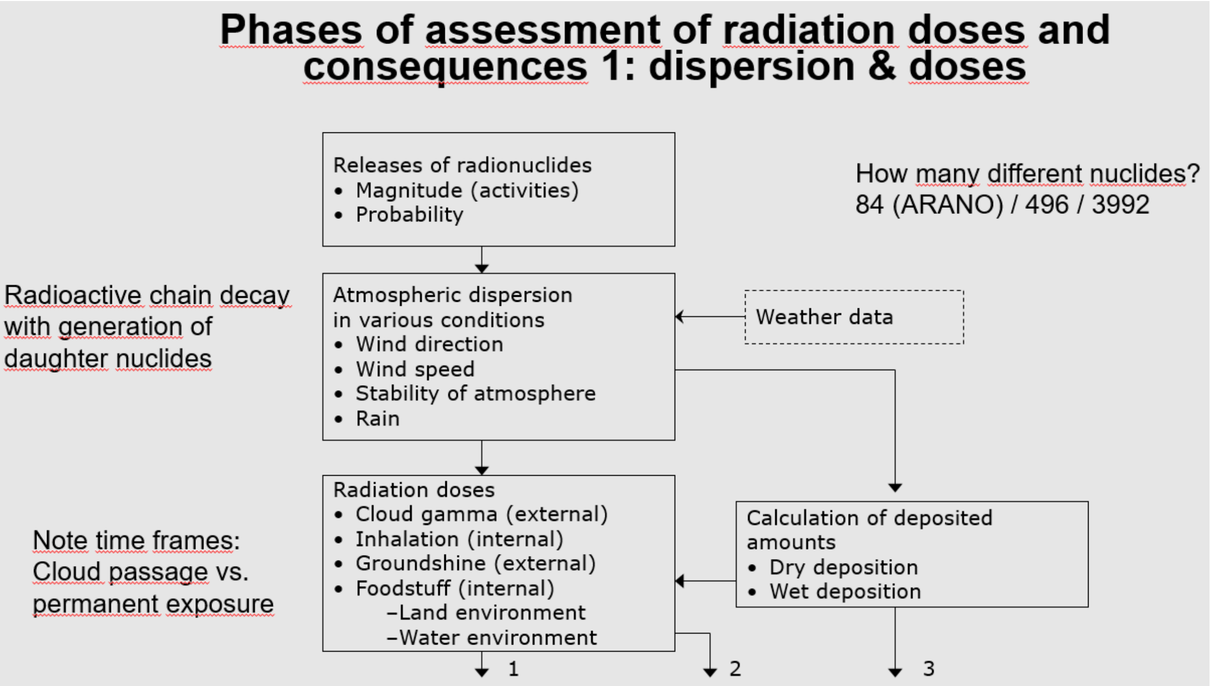
- ARANO (VTT): Gaussian dispersion with internal & external doses, countermeasures
- VALMA (VTT): Dispersion based on 3D trajectories from NWP data
- MACCS (NRC): MELCOR accident consequence code system
- RASCAL (NRC): Radiological Assessment System for Consequence AnaLysis

Suitability of the codes for level 3 PSA of an SMR plant depends somewhat on how well the considerations of Ch. 8.4 (source term), Ch. 9.2 (near-field dispersion) and Ch. 10.2 (dose pathways) are met by the code.

Three local scale consequence assessment models available for SMR EPZ evaluation are briefly introduced here (based on description by J Rossi). ARANO was developed and is owned by VTT, but it was not made for commercial distribution. The other codes (MACCS and RASCAL) are from NRC. VTT has the license to use MACCS, but not for the up-to-date version of RASCAL (5000 $/a). The codes cannot be given further to a third party. The fourth code, VALMA, was developed at VTT, and is best suited for medium and long range (more than appr. 5 km).

ARANO (Assessment of Risks of Accidents and Normal Operation) was developed at VTT (Technical Research Centre of Finland) in the 1970’s and it was initially used for NPP siting studies. The code has been used to estimate effectiveness of different countermeasures. Especially in recent years ARANO has been used by VTT to support STUK in various safety assessments related with construction of a new NPP, dismantling a research reactor, as well as the spent fuel final disposal site.

The working principles of the ARANO code are shown, from initial data to probabilistic consequence results, in the following flow charts. As ARANO is a PSA level 3 code, the chart also gives some idea of PSA level 3 in general.



Phases of assessment of radiation doses and consequences 1: dispersion & doses

Releases of radionuclides
- Magnitude (activities)
- Probability

How many different nuclides?
84 (ARANO) / 496 / 3992

Radioactive chain decay with generation of daughter nuclides

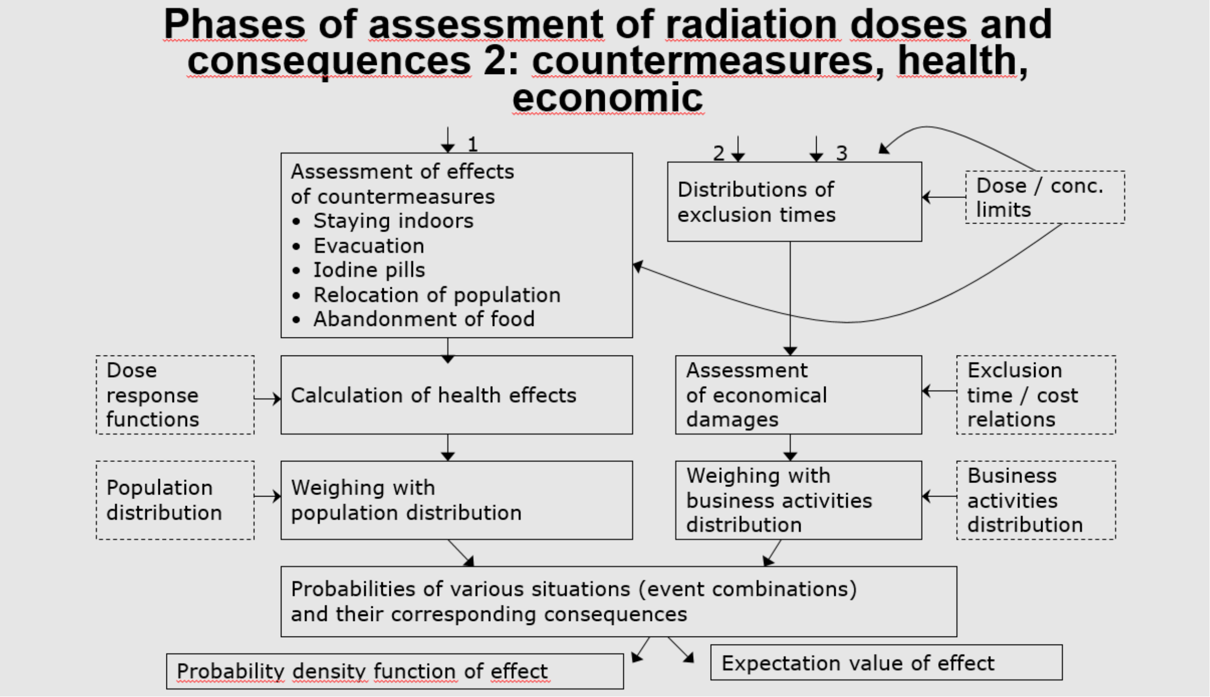
Atmospheric dispersion in various conditions
- Wind direction
- Wind speed
- Stability of atmosphere
- Rain

Weather data

Note time frames:
Cloud passage vs. permanent exposure

Radiation doses
- Cloud gamma (external)
- Inhalation (internal)
- Groundshine (external)
- Foodstuff (internal)
  - –Land environment
  - –Water environment

Calculation of deposited amounts
- Dry deposition
- Wet deposition



Phases of assessment of radiation doses and consequences 2: countermeasures, health, economic

Assessment of effects of countermeasures
- Staying indoors
- Evacuation
- Iodine pills
- Relocation of population
- Abandonment of food

Distributions of exclusion times

Dose / conc. limits

Dose response functions

Calculation of health effects

Assessment of economical damages

Exclusion time / cost relations

Population distribution

Weighing with population distribution

Weighing with business activities distribution

Business activities distribution

Probabilities of various situations (event combinations) and their corresponding consequences

Probability density function of effect

Expectation value of effect

Having simple straight-line dispersion with Gaussian lateral direction, ARANO calculates dispersion and doses fast enough to allow for a number of different source terms, thousands of weather case trials and also fast and easy change of initial parameter values for sensitivity / uncertainty analyses. Those would be harder to do with advanced codes like the SILAM-VALMA system of dispersion and dose assessment, which is more tailored towards high-fidelity predictions of a single emergency (accident) case. Some parameter values important for sensitivity are listed in the following. For example, Hummel et al. (2020) considered a fixed delay of 6 h, but three different release durations (1, 36 and 72 h).

- Delay after SCRAM: It is valuable to contain the release for as long as possible before reaching the atmosphere, as the delay time allows many short-lived but radiologically dangerous nuclides to decrease in activity by decay.
- Duration of release: It is likewise usually good to spread the entering into the atmosphere to a long time span, as that will increase the spread and thus dilution in two ways: in the longitudinal direction when being carried forward by the wind speed, and also in lateral direction because of wind meandering.
- Effective release height: A low release will have worst consequences in the short-range area, whereas a high release (because of a chimney or other cloud rise) will be spread more pronouncedly to longer range. A high release plume may actually touch the ground only relatively far from the source in stable atmospheric conditions, resulting in cloudshine only near the source.

WinMACCS (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. WinMACCS was developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission (NRC). WinMACCS is the code used by the NRC to support Level-3 PRAs. It is currently used in a study of risk-informed emergency response guidance in the U.S. and in an NRC study, the State of the Art Reactor Consequence Analysis (SOARCA), to assess the safety of existing power plants.

ARANO and MACCS treat:

- Atmospheric transport and deposition onto the ground
- In addition to a single dispersion case, also statistical effect of variability in weather
- Dose pathways for cloudshine, groundshine, inhalation, ingestion, and deposition onto skin
- Protective actions during emergency, intermediate, and long-term phases

ARANO and MACCS calculate offsite consequences:

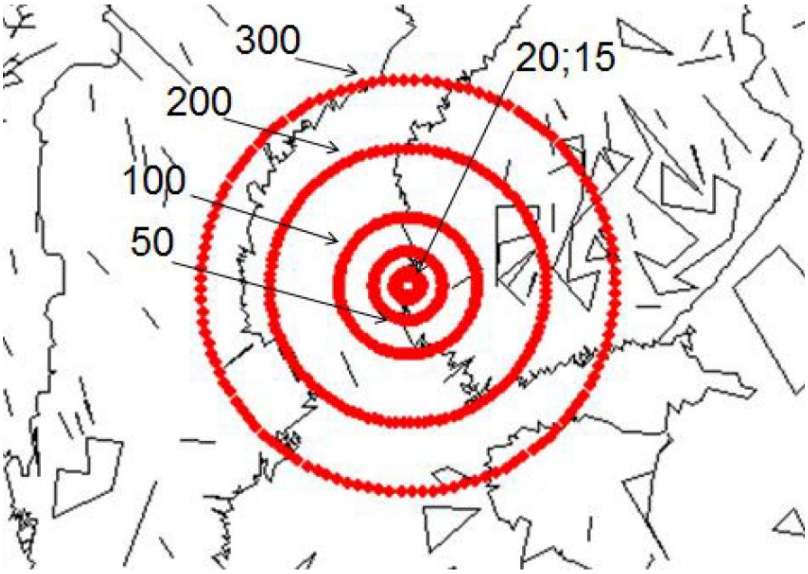
- Doses to individual and population
- Health effects
- Economic costs
- Land contamination

The VALMA model

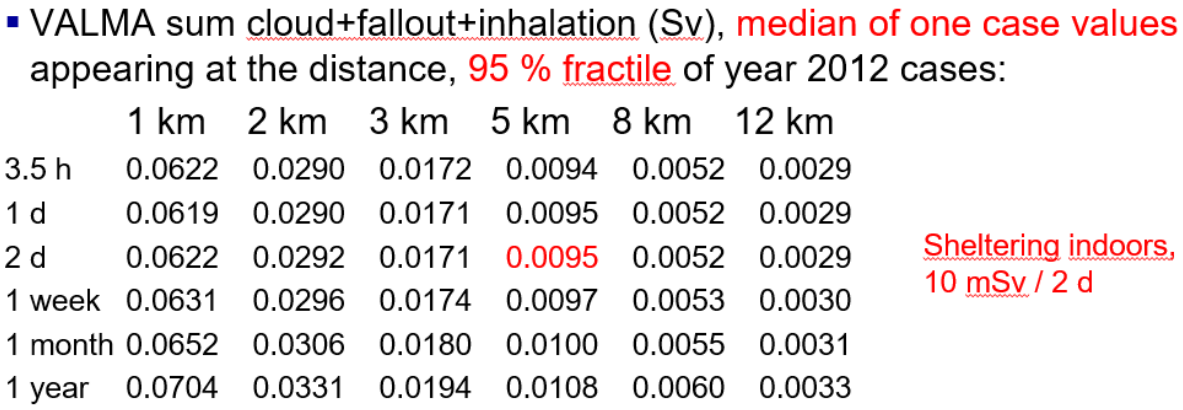
VALMA is a dispersion and dose assessment code for accidental atmospheric radioactive releases [Ilvonen, 2002]. It was developed at VTT in late 1990's and its main purpose was to serve as an emergency preparedness tool for radiation safety authorities (STUK in Finland). In such use, it is essential to produce predictions of concentrations, depositions, dose rates and doses in a reasonably short time to enable possible rapid countermeasures. It is not possible to perform CFD-like calculations that may last hours or days. Furthermore, it is possible that the best existing weather data cannot be received due to e.g. increased web traffic. For this reason, VALMA was made flexible enough to work with many kinds of weather data, starting from single-point measurements at the weather mast of an NPP (or several masts) and ending with Monte Carlo particles (even a limited number) that can be calculated, based on NWP models, with the SILAM dispersion model at FMI. Regardless of the source of weather data, VALMA offers the flexibility to calculate with changing source term estimates, including released nuclide inventory and the temporal and height distributions of different nuclides. It is also easy to set the spatial and temporal grids and to view the Lagrangian trajectories and dozens of result quantities on map or as temporal trends at chosen locations.

In short, VALMA works by dividing the release into a finite number of 'packets' or 'puffs', each of which corresponds to a 'slot' in time and release height. For each packet, VALMA either computes or receives from SILAM a possibly winding central trajectory, which the packet will follow according to available wind information. VALMA follows each packet along the trajectory and calculates its spread, chain decay and deposition scavenging at the same time. VALMA calculates dozens of radiologically interesting quantities, like concentrations, depositions, dose rates and doses via different exposure pathways, together with their time derivatives and integrals. In contrast to an Eulerian dispersion model, VALMA uses a grid only to represent and accumulate the result quantities, not for calculating them. Cooling time between reactor SCRAM and start of the release can be calculated by VALMA. Approximately 100 important nuclides are used.

Even when VALMA uses considerably more CPU time per one calculation case than ARANO or MACCS, it has been successfully used for PSA level 3 studies. For example, Ilvonen (2018) used VALMA to determine EPZ distances around a hypothetical NuScale plant that would be sited in Olkiluoto (and not considering the LLWRs already there). Receptor points were set at several to-be-tested distances in all directions (360 degrees). As there is no such thing as a ‘plume centerline’ in VALMA, one must instead look at all calculated dose values at a distance. Then it is possible to pick from the values certain statistics like mean, median or maximum value. Long-range receptor points, set in the way described above, are illustrated in the figure (from VALMA GUI) below.



The table below, which is Table 6 reproduced from Ilvonen (2018), shows VALMA-calculated total effective doses for various distances and integration times, when considering the 95 % fractile of all weather cases and the median value of the case’s dose values appearing at the distance. (95 % means that only 5 % of weather cases lead to higher doses.) As the dose limit for sheltering indoors is 10 mSv in 2 days in Finland, we see that sheltering could be needed up to 5 km distance. This result depends on the choices of the describing statistic (median or other) and the confidence level (95 % or other). Finally, it must be mentioned here that 5 km is pretty much, and is a result of using just one, extremely conservative, source term in this example.



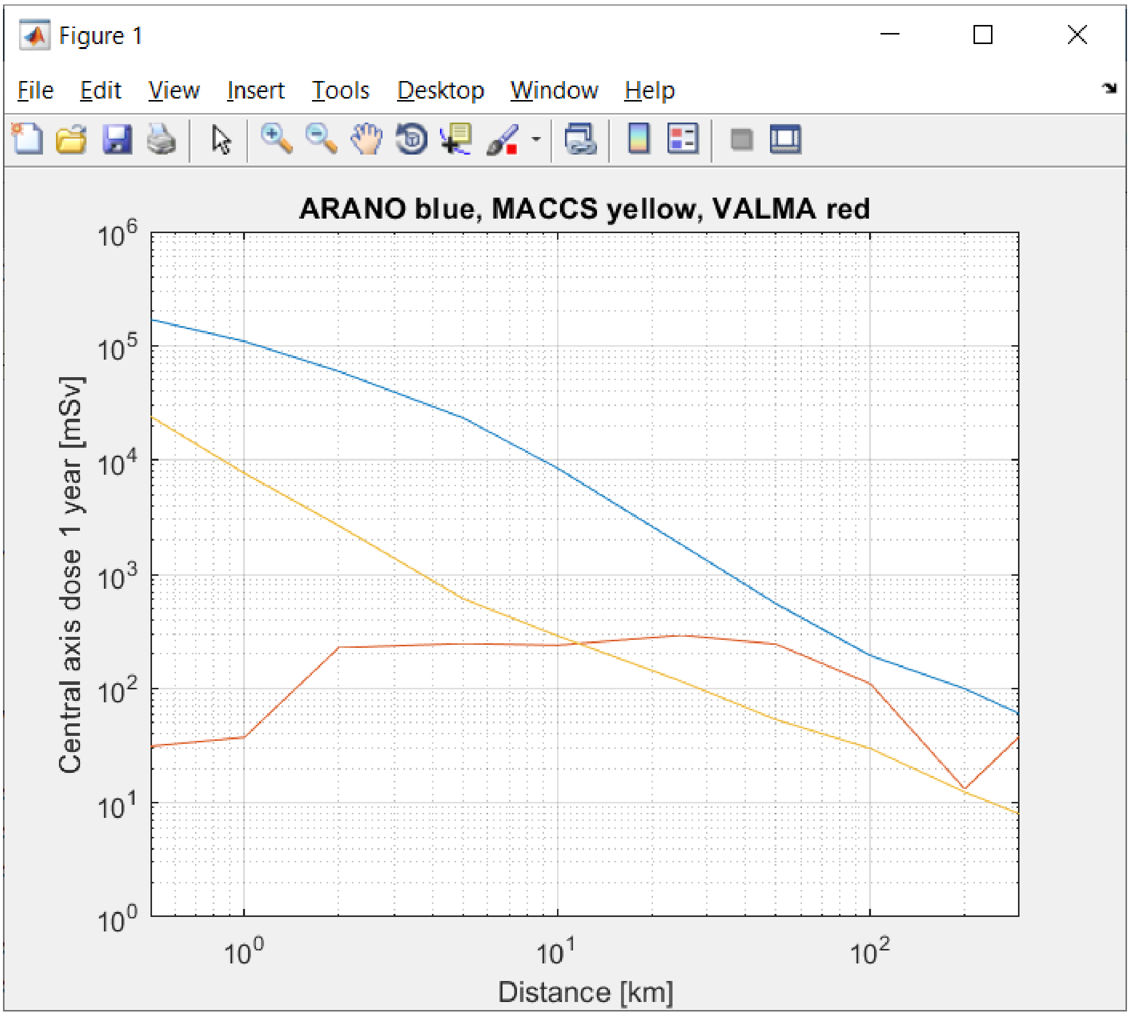
- VALMA sum cloud+fallout+inhalation (Sv), median of one case values appearing at the distance, 95 % fractile of year 2012 cases:

	1 km	2 km	3 km	5 km	8 km	12 km
3.5 h	0.0622	0.0290	0.0172	0.0094	0.0052	0.0029
1 d	0.0619	0.0290	0.0171	0.0095	0.0052	0.0029
2 d	0.0622	0.0292	0.0171	0.0095	0.0052	0.0029
1 week	0.0631	0.0296	0.0174	0.0097	0.0053	0.0030
1 month	0.0652	0.0306	0.0180	0.0100	0.0055	0.0031
1 year	0.0704	0.0331	0.0194	0.0108	0.0060	0.0033

Sheltering indoors,
10 mSv / 2 d

For extensive probabilistic assessments, the straight-line Gaussian ARANO could rather be considered than VALMA, because it calculates much faster than VALMA, and when the distances from source are in the near range (< 10 km), the simple weather description of ARANO should be well sufficient. This is true particularly when the question is not about a single dispersion case but rather a vast number of them, and the important results come from the statistics.

Comparison of the above-mentioned dose assessment codes have been performed by e.g. Rossi & Ilvonen (2019). Another comparison from a 2020 international benchmark (BARCO) is shown here below. The source was a large release from a severe accident at a European large NPP. VALMA had 3D NWP-based weather data from the SILAM dispersion model, run by FMI, whereas ARANO and MACCS used one-point weather. Comparison shows that ARANO and MACCS calculated very similar decrease of dose vs. distance, but ARANO results were one order of magnitude higher. Other comparisons have resulted also in MACCS calculating higher dose than ARANO. However, one order of magnitude difference can still be considered quite good for offsite doses. VALMA predictions are between the other two codes, except that a mistake with initial data caused a ‘gap’ at distances < 5 km. Furthermore, the maximum of VALMA doses (note 1 year integration time) appeared as far as 25 km from the source, which was apparently caused by rain in the SILAM data at that distance, causing wet deposition.



RASCAL

The RASCAL (Radiological Assessment System for Consequence AnaLysis) code is a tool used by the Protective Measures Team in the U.S. Nuclear Regulatory Commission's (NRC's) Operations Center for making independent dose and consequence projections during radiological incidents and emergencies. RASCAL evaluates atmospheric releases from nuclear power plants, spent fuel storage pools and casks, fuel cycle facilities, and radioactive material handling facilities.

RASCAL consists of several modules:

- The first module calculates the time-dependent atmospheric release source term. The atmospheric release source term is the rate at which radioactive material is released to the environment. It also includes other information that defines how the release takes place.
- The second and third modules perform the atmospheric transport, dispersion, and deposition calculations and the dose calculations.
- The fourth module is used to create the meteorological data file used by the atmospheric transport, dispersion, and deposition modules.
- The fifth module is used for intermediate-phase dose calculations based on field measurements.

In the IAEA CRP I31029 (SMR EPZ determination), at least the following dispersion / dose assessment codes have been mentioned / used:

- PC COSYMA
- PACE
- AERMOD, ARCON96, MACCS2, DIFOUT, PAVAN
- CALPUFF
- MACCS

MANCINI et al. (2014) have used the code sequence MAAP-RADTRAD-RODOS-EMERSIM.

MACCS seems to be the main PSA level 3 code in the discussion between the US industry and NRC. In their feedback questions to NEI (2013) SMR EPZ methodology, the NRC was asking about MACCS use:

The paper states that the MELCOR Accident Consequence Code System (MACCS) code is an appropriate tool to calculate consequences for the analyses. The paper also states that insights from the NRC's use of MACCS for the SOARCA will be used to inform use of the code for this purpose.

a. *Considering that MACCS has not previously been used for SMR analyses to support EPZ sizing, will NEI provide more information or guidance on the use of the MACCS code for this purpose? Topics that could be different for this proposed use of MACCS are the determination of the basis for code input and model assumptions (including appropriate nodalization of the near-field), sources of information for input, use of conservatism, addressing uncertainty, and input and assumptions for the local area and population.*
b. *Which information from SOARCA is proposed to be used, and how will it be used?*
c. *Will a test case or pilot case be provided for demonstrating how MACCS and SOARCA information would be applied in a realistic situation?*

Talabi & Fischbeck (2016) emphasize the advances made in modelling techniques since 1978 assessment: ‘Significant advances in analytical techniques including computational fluid dynamics and stochastic methods (e.g. Monte Carlo sampling with software)’ mean that ‘overly conservative assumption of plume straight line trajectory can now be more accurately evaluated’.

There are numerous possible sources of differences between PSA level 3 results and thus also EPZ size determination for a certain type of NPP, even when the same code is used, many of them related to the so-called ‘user effect’:

- Site-specific weather data and how it is adapted for use in the code; also kind of weather data (NWP-based or mast-measured; quantities measured)
- Other site data, like surface roughness, resuspension
- Adaptation of atmospheric source terms, e.g. from MELCOR calculations, in the level 3 code: How their details (like temporal behaviour of different nuclides’ release rates) are taken into MACCS (or other code) input.
- Also, when using the GSR Part 7 generic dose criteria, there are many options / possible pitfalls when interpreting them for MACCS (or other code) input, like integration time settings, choice of nuclides, absorbed vs. effective dose, etc.
- Shielding factors, breathing rates, exposed skin area etc. all have to be set to appropriate values (if the setting is available in the code).

10.4 Acceptable health effects and other consequences

The dose limits for offsite public are based on the allowable adverse health impact of radiation. Exceeding a limit of mitigating (protective) actions means that the actions should be performed. After the protective countermeasures, a certain remaining residual dose may have to accepted.

The modelling of radiation-induced health effects is usually divided in three types of effects: early (acute), late (cancer) and genetic effects. The acute effects include e.g. radiation sickness, various organ damages, and possible death, which usually results in the dose range 1…6 Sv (cf. Figure 11). Acute health effects appear after a threshold value of radiation dose is exceeded during a short time of exposure (usually considered to be less than 1 week or 1 month). Late health effects are generally assumed directly proportional to the radiation dose, with no threshold. The 'traditionally' used coefficient (according to ICRP) is 5 %, i.e. the total number of cancers = 0.05 x population dose (manSv).

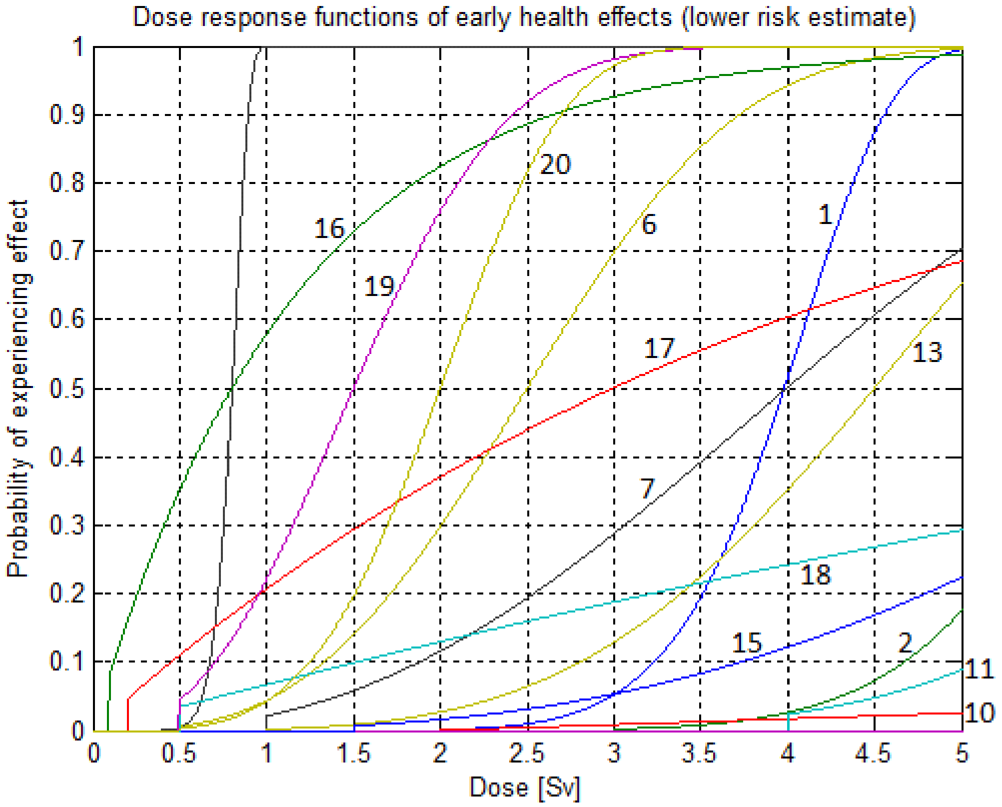


Figure 11. Dose response functions (probability of health effect as a function of dose) of early health effects in VALMA, as a graphical representation <u>in the interval 0…5 Sv</u> of the appropriate organ-specific dose (to be chosen appropriately for each effect). For the numbering of effects, refer to VTT-R-00885-18. The functions plotted here are for the lower estimate of risk. SMR offsite doses could drastically reduce potential health effects just by keeping under the threshold values.

The characteristic of acute health effects of having thresholds above which they become significant has been mentioned as a potential basis for SMR reduced EPZ sizes. Just by reducing the offsite dose to a certain fraction (e.g. according to the ratio of MW_{th}) of a large NPP, we might reduce the corresponding probability of experiencing a health impact (at the same distance & time point) by far more than the ratio of doses, if the dose got under the threshold. Note that the dose response functions of Fig. 11 have two kinds of successive thresholds: One 'total' threshold, under which the probability has only zero values, and above that is another, 'sigmoid type' threshold, at which the probability rises more or less steeply from near-zero to one.

The generally accepted policy in mitigation of adverse health effects induced by radiation dose has two main points:

- No individual person should experience any acute effects from radiation.
- The increased risk of stochastic effects (i.e. cancer in practice) should be as low as reasonably achievable (the ALARA principle).

IAEA General Safety Guide GSG-2 (Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency) from 2011 (120 pages) considers the dose criteria of response actions and other intervention levels:

- Generic criteria, expressed numerically in terms of radiation dose
- Basis for operational levels to decide about protective actions
- Lessons learnt from responses to past emergencies
- Plain language explanation of the criteria for the public
- Response actions, projected / received dose, emergency workers, operational criteria, dosimetric quantities
- Operational intervention levels (OILs):
    - Deposition
    - Individual contamination
    - Contamination of food, milk and water
- Emergency action levels (EALs); general / site area / facility

The newer reference for the list above is the IAEA EPR-NPP_PPA (2013).

The EPR-NPP_PPA (IAEA 2013) contains in Appendix III some advice for putting the radiological health hazards in the right perspective. Four color-coded levels are used:

- Possibly dangerous to health (red): There is a possibility of severe deterministic health effects (life threatening, or permanent injury that reduces the quality of life). Also the small possibility of an observable increase in cancer.
- Health concerns (orange): The danger to health is very low. However, there can be (a) the small possible risk to fetus, and (b) the small possible increase in cancers.
- Provisionally safe (yellow): Safe for all members of the public, including the most sensitive (e.g. children and pregnant women) and there are no hazards if the specified limitations (e.g. ingestion) are followed.
- Safe (green): This meets international standards and is therefore safe for all members of the public, including the most sensitive. There will not be any severe deterministic effects or an observable increase of cancer, even in a very large exposed group. The risk of radiation induced cancers is too low to justify taking any action at all.

The most decisive health effects are those due to the radiation dose in certain sensitive target organs, like the thyroid and the unborn fetus.

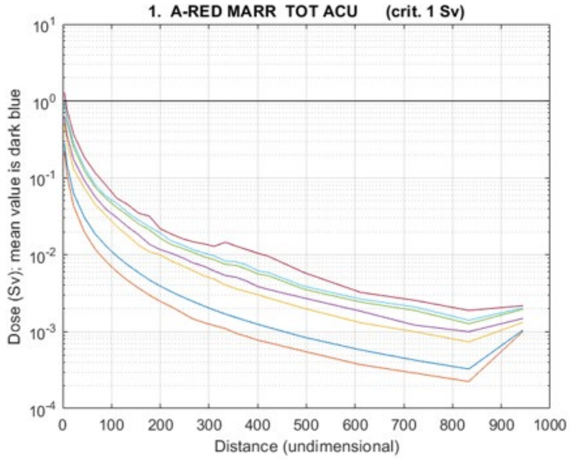
10.5 Sample MACCS & ARANO calculations: dose vs. distance

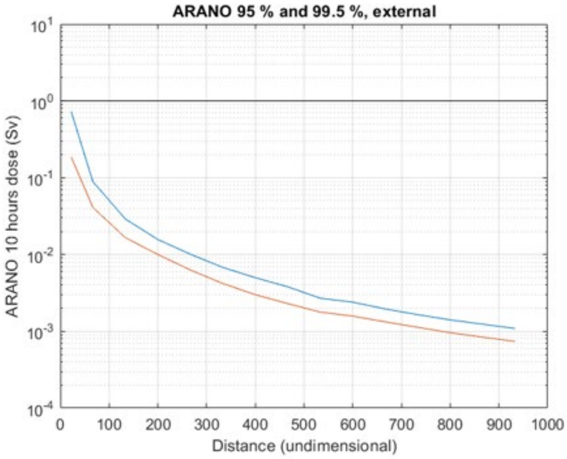
In the 2020 RCM3 meeting of the IAEA CRP I31029 (SMR EPZ), VTT presented some MACCS and ARANO calculations for a hypothetical SMART (Korea) nuclear power plant sited at Olkiluoto, using weather data measured at Olkiuoto mast. The EPZ determination was based on the following definitions of exposure pathways 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).

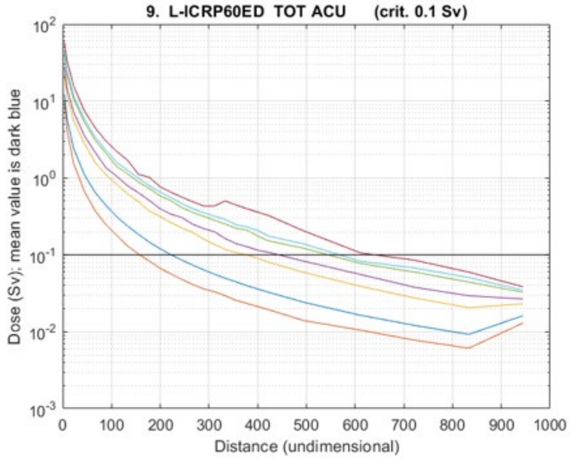
	Exposure type	Exposure duration	Organ	Dose type
1	External	10 hours	Red marrow	Absorbed
2	External	10 hours	Skin	Absorbed
3	Internal	30 days	Red marrow	Absorbed (Z≥90)
4	Internal	30 days	Red marrow	Absorbed (Z≤89)
5	Internal	30 days	Thyroid	Absorbed
6	Internal	30 days	Lungs	Absorbed
7	Internal	30 days	Colon	Absorbed
8	Internal+External	7 days	Thyroid	Equivalent
9	Internal+External	7 days	Effective dose	Effective
10	Internal+External	First year	Effective dose	Effective

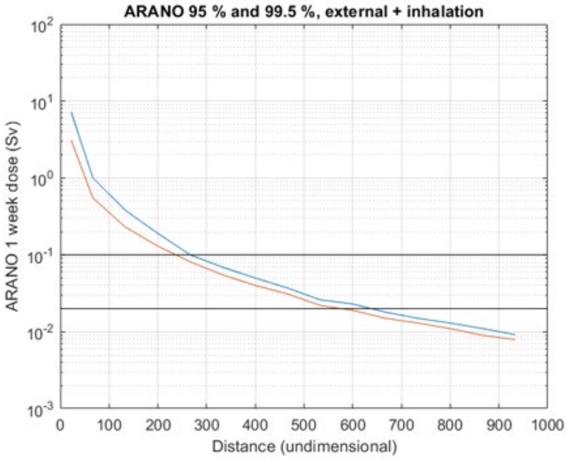
MACCS and ARANO results in the following figure are for criterion 1 in upper row (Deterministic effects, external exposure, integration time 10 hours, red marrow) and criterion 9 in lower row (Stochastic effects, external + internal, 7 days integration, effective dose). MACCS fractiles (or other statistics), from top to bottom curve, are Peak, 99.5 %, 99 %, 95 %, 90 %, Mean, and 50 %. From ARANO, only the 99.5 % and 95 % fractiles are shown. For example, Hummel et al. (2020) use ‘P90’ or 90 % fractile / percentile. Note that the distances were shown as undimensional only, because this result was calculated from only one certain severe accident release (so showing EPZ results from that would be misleading before more releases are considered).

To compare with each other are the MACCS purple curve with ARANO red (for 95 % fractile) and MACCS light blue with ARANO blue (for 99.5 % fractile). For criterion 1 (integration 10 h) the models are in good harmony, but for criterion 9 (integration 7 d) ARANO has calculated somewhat lower doses. The use of more than model is valuable for catching possible errors, including also user error.









EPZ distances at a given dose level (dose criterion for action) & probability level are shown in the following table (undimensional distance shown). Each distance entry in the table was picked conservatively as the upper bound of MACCS distance interval. Columns of the table correspond to the GSR Part 7 generic dose criteria (1. Ext 10 h red marr, 2. Ext 10 h skin, 4. Int 30 d red marr, 5. Int 30 d thyroid, 6. Int 30 d lungs, 7. Int 30 d colon, 8. Tot 7 d thyroid, 9. Tot 7 d effective, 10. Tot 1 a effective). Each row of the table corresponds to MACCS dose vs. distance result for the shown percentile. The table shows that we are getting a spectrum of EPZ distances (differing more than 300-fold), depending on dose pathway and fractile. When the chosen fractile is 'squeezed' up towards higher confidence, the distance correspondingly must be increased.

	1	2	4	5	6	7	8	9	10
mean	3	44	9	133	3	3	722	244	289
50 %	3	22	3	89	3	3	611	178	222
90 %	3	67	22	244	9	3	944	400	494
95 %	3	89	22	289	9	3	-1	494	611
99 %	3	111	44	356	22	9	-1	611	722
99.5	9	111	44	378	22	9	-1	611	722
peak	9	133	44	494	22	9	-1	722	833

With the dose results available, it is easy to do a simple calculation experiment and divide thermal power / accident source term / all doses by 10 (only an approximate procedure) and see what happens to the EPZ distances at various dose criteria and fractiles. The recalculated table below shows that decreasing doses to 1/10 may only reduce the distances to approximately ¼.

	1	2	4	5	6	7	8	9	10
mean	3	3	3	22	3	3	200	67	67
50 %	3	3	3	22	3	3	133	44	67
90 %	3	9	3	67	3	3	333	111	133
95 %	3	9	3	67	3	3	400	133	156
99 %	3	22	3	89	3	3	494	156	178
99.5	3	22	3	89	3	3	494	156	200
peak	3	22	3	111	3	3	611	200	200

11. How to use the PSA results?

One central question in the determination of EPZ size for SMR is whether the result is 'tailor-made' for a certain plant type and/or site, or if it can be applied more broadly to similar SMR plants with the same kind of technology and power level and possibly even at different sites. The latter would be a generic, prescriptive and probably conservative option, but would then also mean less work in the licensing phase for both applicant and regulator.

However, just theoretically and using lot of labour, the widely applicable results of EPZ size could also be generated by full-scope PSA covering several (somewhat similar) plant types and several sites (with at least somewhat similar external events), and then 'enveloping' the results according to the worst outcomes. A similar thinking in 'lighter form' was actually behind the ESPA (early site permit application) of TVA (Tennessee Valley Authority) for the Clinch River site in May 2016:

'This application is for the Clinch River Nuclear (CRN) Site in Oak Ridge, Tennessee, and is based on a plant parameter envelope encompassing the light-water small modular reactors (SMR) currently under development in the United States by BWX Technologies, Holtec, NuScale Power, and Westinghouse.'

Theoretically, a PSA based EPZ methodology could have use both offline and online. The former means the 'usual' full-scope PSA procedure suggested in this work to produce right-sized EPZ from plant design and site-specific information. The online application would use the current accident information to determine which branches (accident sequences) might probably be ongoing and reduce the PSA level 3 outcomes to those. (Note: The VTT-made ROSA code, used at Olkiluoto, has event trees for user choices and then predicts offsite doses using currently measured weather.)

Mancini et al. (2014) categorize the principal ways to use the PSA level 3 information of doses & frequencies in the following four options (different rationales):

- Consequences: The most usual way of working at VTT has been to look at the dose vs. distance curve, calculated for a specified source term, but using many years of statistical weather data, then each value of dose chosen from e.g. the 95 % or 99.5 % fractile of all values at the distance. This kind of curve (see Fig. 7) shows then directly the distances at which certain dose limits are exceeded.
- Probability (frequency): Mancini et al. (2014) have plotted curves using 'opposite' thinking: They plot vs. distance the frequency of exceeding a dose limit. If this frequency is below a specified value, then the distance is considered to be on the 'safe side' with regard to the limit.
- Risk (risk = consequence x probability): Looking at both the doses and frequencies at the same time. For a worse consequence, the frequency must be correspondingly lower. Then the risk is at the same level for minor incidents and major accidents. However, due to the public perception of risks (aversion of large accidents), the probability goal is usually set still lower for large accidents (see e.g. Figure 17.22 of Rahn et al. 1984: Farmer-type lines with different slopes). For example, in UK and Argentina risk is part of acceptance criteria.
- Cost-effectiveness: The costs of maintaining emergency preparedness arrangements within a certain zone can be compared with the expectation value of accident offsite recovery costs. A larger EPZ will cost more, but also reduce the mitigation costs, should an accident happen. However, there are many other considerations at the same time, for example usually no individual should bear untolerable risk. INL (2014) discussed cost-benefit assessment of SMR EPZ sizing.

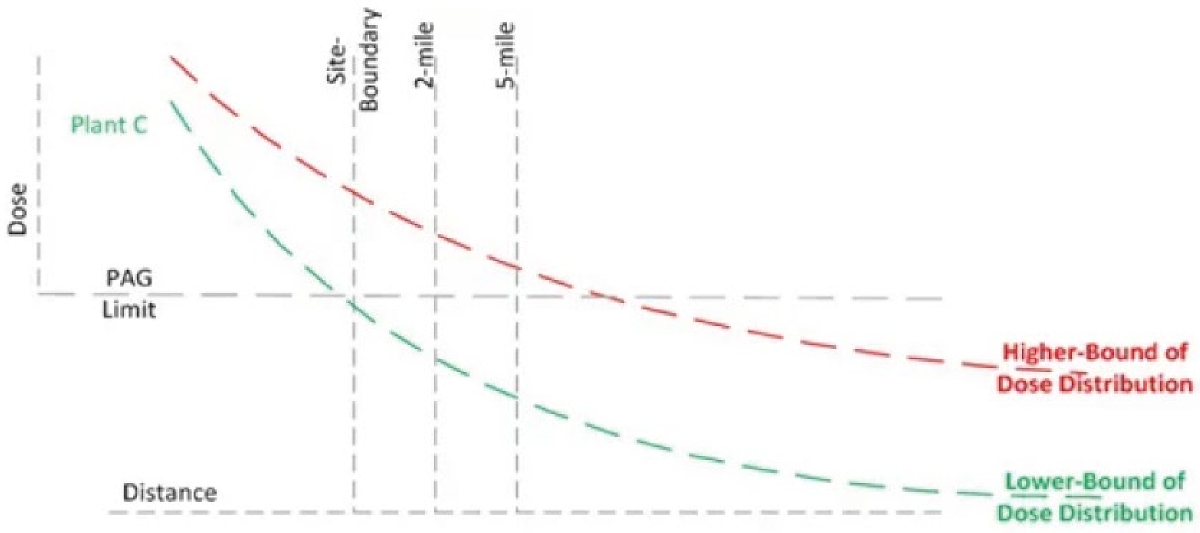


Figure 13. Appearance of dose vs. distance curves, when the dose at each distance has a probability distribution (shown lower and higher bounds). Source of Figure: Pittsburgh Technical, Determining the appropriate EPZ for SMR, https://www.pit-tech.com/.

Various protective countermeasures usually have their criteria of onset defined as individual radiation dose to the members of the offsite population during a certain time period from the start of the nuclear or radiological emergency or from the start of an atmospheric radioactive release. So basically to define the appropriate extent of the various zones around the nuclear plant, we should understand the following prerequisites:

- The meaning of the different zones, i.e. what kind of protective actions are assumed to possibly take place there.
- What is the criterion, in most cases the expected radiation dose during a certain time period, for taking each protective action.
- Which actions would be necessary at incremental distances from the release source – to find, for each protective action, the maximum distance where it could be necessary to perform.
- The probability of each action to be required at each calculation distance. For one certain fixed radioactive source term to the atmosphere, the probability results from varying weather conditions, part of which lead to a lower dose and others to a higher dose than the criterion at that distance. Note that this is a conditional probability, assuming (e.g. for a severe accident) that core damage and containment failure actually occurred.

It is very difficult to give unambiguous recommendations on the EPZ size based on the dose vs. distance results, but some considerations are given here (cf. Chapter ‘Protective countermeasures’ for IAEA / STUK criteria). An illuminating example here is from GENXFIN work in 2017 (VTT-R-00651-18, Ilvonen 2018), shown also above in Ch. 10.3 in connection with the VALMA offsite dose assessment code. The VALMA code with 3D weather was used, and so at each distance, several different dose values resulted. From those values in one calculation case, either mean, median or maximum could be chosen to represent the one and single dose at the distance. Then, from the statistics of all calculation cases (the CCDF curve, or complementary cumulative density function) one can choose a suitable fractile, like 95 % or 99.5 % (meaning the probability that the dose will be lower than the chosen value). Depending on these simple choices, an illustrative variety of possible maximum distances for protective measures was obtained:

- If we assume the need for sheltering because of 10 mSv in 2 days, the radius of this effect could be 5 km, 8 km, 10 km or even 12 km, depending on the choices.

- If we assume the need for evacuation because of 20 mSv in 1 week, the radius of this effect could be 2.5 km, 5 km or 8 km, depending on the choices.
- If we assume the need for relocation because of 30 mSv in 1 month, the radius of this effect could be 2 km, 4 km or 5 km, depending on the choices.

Some potentially valuable guidelines for applying the ’dose at distance’ results to determine the final extent of EPZ:

- The distance value (shortest distance where a certain dose criterion is not exceeded) will depend on dose pathway and choice of percentile (e.g. 95 % or 99.5 % of all weather trials remaining lower than the dose criterion, at that distance).
- The probability of a typical ‘worst-case’ source term (from PSA2) is very low.
- It is best not to present dose values or distances from a single source term calculation; various source terms should be analysed at PSA level 3.
- It remains for the regulator to decide the level of confidence (when setting distance etc): Which consequence to accept at which probability?
- Not just the size of EPZ, but also graded approach to protective measures could be applied, meaning that a subset of all existing measures may be considered and prepared for.

Views may be different on different sides. According to NEI (2013), representing US nuclear industry side (as already discussed above in this report) ‘The size of the EPZ should be such that consequences from more probable, less severe accidents would not exceed the PAGs outside the EPZ, and should also provide for substantial reduction in early severe health effects in the event of less probable, more severe accidents.’

11.1 Risk-based approach

In a risk-based approach, the acceptable frequency must go down with increasing dose consequences. A dose received at a certain distance is acceptable, if it is below the dose limit. In addition, it can be acceptable also if above limit, but the frequency of such doses occurring is sufficiently low. If not acceptable, the design should decrease either doses or frequencies or both. If acceptable outcome at the distance is not achieved, there will have to be sufficient emergency / mitigation arrangements there. An old, but illuminative treatment of risk is given e.g. by Rahn et al. (1984), pages 748-762.

Mancini et al. (2014) state four possible rationales for the emergency management criteria to be based on: risk, probability, cost-effectiveness, and consequences. They have chosen a risk-informed approach, which ‘links the emergency management with the safety level of the NPP’. Some successive steps & highlights from their work are the following:

- Step 1: Choose release scenarios A_i with their frequencies of occurrence f_i
- Accident re-categorization is performed to avoid excessive number of scenarios. This is a traditional step in PSA when moving from one level to the next.
- Step 2: Deterministic evaluation of consequences (best estimate, with complete set of meteorological conditions; in practice, it seems that only rainy day / stable day have been used) including all sequences with SA, producing for each A_i curves dose vs. distance: D = f(x) which generally have a different shape for different releases.
- Step 3: Choose dose limit D*, like in the usual consequence-oriented approach, suggested e.g. by the US PAG manuals, agreed with regulator
- Step 4: Choose frequency limit f* (1e-7 chosen according to NRC documents; including the frequencies of release scenarios) which will be used for frequencies of exceeding the dose limit
- Step 5: Choose distances up to which the dose limit D* is exceeded more frequently than the frequency limit f*; at those distances emergency planning will be needed (see Fig. 14).

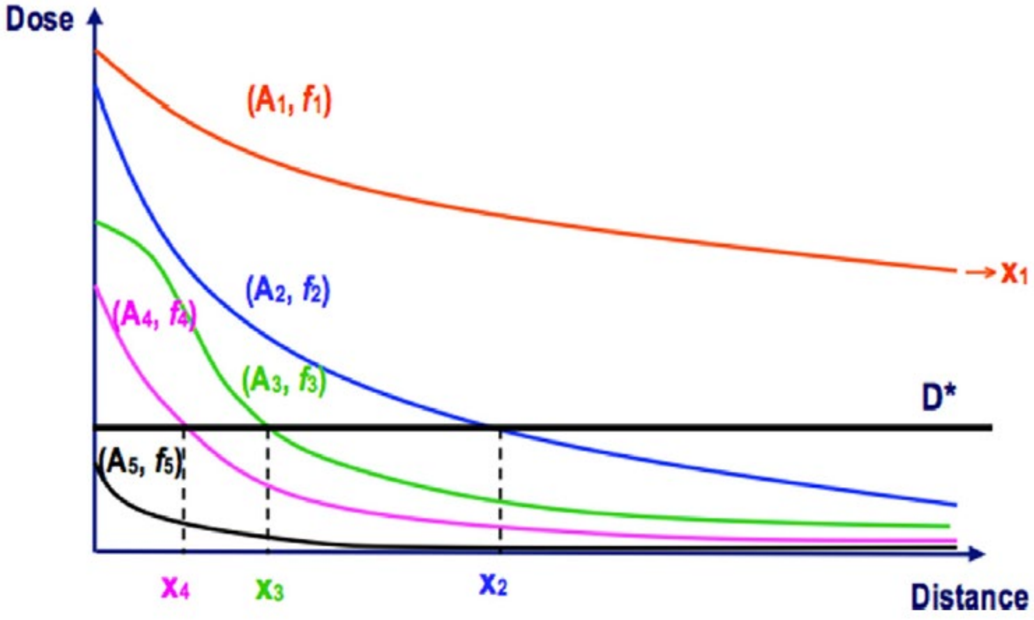


Figure 14. Solving for distances x_i of exceeding the dose limit D_i by Mancini et al. (2014). In practice, using e.g. ARANO at VTT, the distance where the frequency limit (of exceeding the dose limit) is exceeded could be found more directly by calculating all the combinations of source term (release) and weather condition, and from each resulting D = f(x) curve, adding up the frequencies at each distance.

Mancini et al. (2014) justify their limiting frequency (1e-7 events per reactor year) as ‘general consensus for a meaningful decision-making process’ but controversial. For dose limits, they refer to ICRP (cf. Ch. 4.4 above): e.g. 100 mSv / 30 d, effective dose, and assume (very conservatively) living outdoors and purely local ingestion. For the IRIS SMR plant, the results have either frequency below limit, or dose below limit. However, another trial with 10 mSv limit leads to non-zero distances, like 932 m (rain) or 1735 m (stable day). Mancini et al. (2014) conclude that IRIS could ‘avoid the implementation of countermeasures also for severe accident scenarios, considering that at the very conservative limiting dose of 10 mSv an EMR area can be established in less than 2 km radius’.

The interested reader may refer directly to the following SMR results by Mancini et al. (2014):

- Mancini Table 11: SMR maximum doses for release categories (‘close to reactor boundary’)
- Mancini Table 12: SMR distance of dose limit 10 mSv (release categories R1-R14, rainy day / stable day)
- Mancini Fig. 14: SMR frequency (of dose > 10 mSv) vs. distance; rainy day / stable day
- Mancini Fig. 15: Frequency vs. distance curve with uncertainty distributions in both

It must be noted here that the methodology in the publication by Mancini et al. (2014), which describes the SMR EPZ work concerning IRIS (International Reactor Innovative and Secure), of ENEL (Rome), UPM (Madrid) and PoliMi (Milano), was prior to the publication partially described also in IAEA TECDOC 1652 (IAEA 2010) Annex I, authored partially by PoliMi.

In usual licensing requirements, the acceptable consequences of different plant states (NO, AOO, DBA, DEC) differ according to their expected relative frequencies. Usually worse radiological consequences are accepted for states which are very uncommon (very low frequency), and so normal operation has the strictest dose limits. However, the exact criteria differ for different IAEA member states. In Finland, there are dose limits according to the assumed frequency of the accident (Class 1 or 2 DBA or DEC) - see Ch. 4.5 above.

NEI (2013) states in their 'White paper' on SMR EPZ that 'The industry proposed use of risk information in EPZ size determination is not a risk-based approach. Rather, it is to inform, i.e., a risk-informed approach. A risk-informed approach is a combination of traditional and risk-based approaches through a deliberative process. It balances risk considerations and defense-in-depth.' They also describe Regulatory Guide 1.174 (An approach for using PRA in risk-informed decisions on plant-specific changes to the licensing basis) as specifying the 'use of PRA methods and data in a manner that complements the NRC's deterministic approach and indicates NRC's desire to base its decisions on the results of traditional engineering evaluations, supported by risk insights'. The NEI would also like to see that that the technical basis for EPZ size is 'insights, not just numbers or criteria'.

In their feedback questions to NEI (2013), the NRC wanted to know the proposed probability basis for NEI Criterion c ('substantial reduction in early severe health effects in the event of more severe core melt accidents'; probability of dose exceedance) and asked whether it is probability over 'weather trials, over scenarios, over accident classification (frequent, infrequent, severe), over type (internal, external, low power and shutdown, internal flood, internal fire, other), over release categories, or something else'.

11.2 EPZ considerations other than dose vs. distance

Usually the EPZ is not exactly a circular area around the NPP up to a certain distance (like the bigger EPD and ICPD of Ch. 4.1 are), but rather the exact size and shape of each EPZ is determined by considering many other factors than just the characteristics of the plant:

- Geographical features where the plant is located
- Usually some features 'evident from map', like rivers, railways or main roads, can serve as exact bounds.
- Population areas surrounding the plant
- Practical feasibility of performing the necessary protective countermeasures (e.g. traffic connections)
- In Finland, municipality bounds are taken into consideration to facilitate the work of local rescue units.
- Generally, different countries might deduce different EPZ sizes from the same SMR design, depending on policy factors and public acceptance, in addition to dose criteria.
- According to the IAEA Regulators' Forum EPZ WG (2018), 'The dose reduction factors applied from EPR-NPP_PPA 2013 Appendix I are based on simple assumptions in regards to public behaviour, affecting the effectiveness of actions. These assumptions can vary from state to state, and some states may choose not to incorporate any protective actions into the dose consequence calculations.

Hidayatullah & Subki (2015) claim that 'Compared to traditional NPP sites, there is greater potential for SMRs to be located in countries or regions in a country where basic infrastructure to support facility operation or emergency planning is insufficient or does not exist (e.g. Roads, hospitals, local emergency response capabilities). This is particularly true in very remote regions in the Far North or in areas of very low population density where a project such as a mining, needs to establish stand-alone power infrastructure'.

Somewhat on the contrary, the IAEA SMR Regulators' Forum EPZ WG report has, among other specifically highlighted issues, that 'According to existing IAEA Safety Standards, it would not be appropriate to consider EPZ&D as a design issue (i.e. as being related / influenced by the design safety)'. In other words, regardless of the design, some emergency preparedness should always exist.

As already discussed above in this report, the appropriately sized EPZ should be the basis for a certified offsite all hazards plan, which includes defining emergency action levels (EALs), emergency drills and training, protective action strategies, and a modern public alert system. The decision on all of these is related to the decision on the size of the EPZ. The possibility of needing to expand the response efforts should also be considered. According to Philip Vilar-Welter (Emergency Preparedness Officer of the IAEA), the size of EPZ can ultimately be determined only in connection with integration into the overall protection strategy.

INL (2010) mention also the following factors other than offsite dose consequences that influence EPZ size:

- Relationship to local emergency response needs and capabilities
- Assessment of interfaces with, and requirements of, coordinating agencies (e.g., within the National Response Framework)

Ramana et al. (2013) warn that

- Smaller EPZ (and other weakened requirements) may erode all SMR safety advantages.
- SMR enhanced safety features could be offset by e.g. siting closer to urban areas.

12. Conclusions

A literature study of SMR emergency zones was performed. The growing international interest in SMR nuclear power plants has included discussion on the required size of their emergency preparedness and response (EPR) distances or zones. Plant providers usually try to justify small zones, whereas the regulators, in many cases, have not made it clear if and with exactly what justification those can be approved during the licensing process. Emergency distances from 1 km to 2 km have been mentioned for e.g. KLT-4S, VBER-300, ABV, SMART and IRIS.

These achievements could be realized taking advantage of the smaller reactor core radioactive inventories and the more advanced safety features of the new plants. However, the possibly smaller EPR distances around new types of nuclear power plants and the possibility to completely do away with EPR arrangements remain controversial.

Ideally, a rigorous analysis of the EPR distances of SMRs should be made based on actual radioactive inventories, modelled DF (leak path decontamination factors) and resulting atmospheric release source terms, as well as a computational assessment of doses and their comparison with international action levels for radiological countermeasures. The main question is how far from an SMR plant the PAZ (precautionary action zone) and UPZ (urgent protective action planning zone) should reach, which is of particular importance if the SMR plants are to be used as a local source of heat for cities and industry.

This literature study task covered emergency planning provided by SMR plant vendors, examples of regulatory policies internationally, and analyses on the topic made by research institutes and consulting firms. Recommendation on the use of level 3 PSA was made based on examples like the Canadian licensing process, and some details of VTT capabilities in that field were given.

Some off-site EPR arrangements should exist, even when the SMR is considered ’inherently safe’.

- EPR exists for the unexpected, unconsidered
- ’EPR is not a design issue’ (IAEA)

For the EPZ sizing problem: How safe is safe enough?

- There are always unacceptable consequences with some very small probability
    - ’Meteorite strikes from outer space and blows the inventory into the air?’
- How to prove anything for first-of-a-kind?

More detailed models are a good thing if they

- Are practically computable (CPU time issue)
- Do not bring too many additional unknowns

The following points (1-4) express the author’s personal view on the SMR EPZ problem:

1. We still need EPZs. We can never get rid of the EPZs, because then we would completely trust the design. Design can always be incomplete, or it could be spoiled by bad quality in construction, operating organization, etc. Meteorite might strike from space, terrorists might attack, not to mention the possibility of military action. Core radioactive inventory means that the potential for damage is there.
2. All of EPR should be more harmonized. The average citizen may wonder why something is safe in one country, but not so in another. Also the differences in EPZs and EPR in general.
3. Better understanding and communication. People should be taught better understanding, like why we try to avoid each mSv during normal operation, but then 1 mSv is actually not so dangerous in an emergency.
4. We should study the case for SMR. We should do rigorous work to determine for SMRs right-sized EPZs, evidently smaller than currently. This involves exact core inventories, conservatively chosen

assumptions of released fractions, and level 3 PSA study of the off-site consequences using real weather of several years.

An attempt to crystallize the 'IAEA point of view' on SMR EPZ (not officially existing in any detailed guide), as it seems apparent in the author's experience of several IAEA meetings, is given here:

Whatever power level, whatever inherent safety, whatever engineered safety systems, but the following facts remain:

- We still have a radiation source, i.e. radioactive fission products.
- We may still encounter completely unexpected situations (as history has shown).
- DiD level 5 should still exist in some way.
- Health hazard with relation to dose (i.e. dose response functions) to humans are the same as always before.
- Same fundamentals of EP&R as before are still valid:
    - GSR Part 7
    - EPR Method TECDOC-953 (2003)
    - EPR NPP PPA Public Protective Actions (2013)
- But the detailed guidance may be different:
    - Optimized, justified, scalable, graded approach (GA)
    - EPZ area, but with only some protective measures planned?
    - Combining with conventional emergency preparedness

Hidayatullah & Subki (2015) say that 'Even if SMR designs are made more inherently safe and to have reduced environmental impact, adequate infrastructure for emergency planning needs to exist from a defence-in-depth perspective to respond to plant emergencies and from a public confidence perspective'.

This report also contains some information on SMRs in general, international (IAEA) safety standards, and international collaboration where VTT participates.

13. Suggestions for future work

In recent years, small modular nuclear reactors (SMRs) have been an increasingly discussed topic internationally and also in Finland. The main reason for this is probably the issue of financing risks. Particularly in western countries, the costs and time schedules of construction projects of traditional big nuclear power plants (NPPs), typically over 1000 MWe, have exceeded those mentioned in the planning phase. A widely known example of this is Olkiluoto-3 in Finland. By dividing the power generation among several smaller modules, pre-fabricated as serial production, it is expected to reduce the financing risks. However, to be economically competitive (economy of mass production, instead of economy of scale) there would have to be a large number of similar units produced.

Another advantage of smaller power is the inherently easier heat transfer: power increases roughly with volume of the core, but heat transfer capability by surface area. This makes it easier to rely on passive safety systems, which may be more reliable and less expensive than the active ones (pumps, AC power etc.) typically needed in larger plants. Examples are relatively large water volumes, heights of water columns and natural circulation.

Smaller power of units is better suited for heat-producing applications, as it is not economical to transfer heat over long distances like electricity. So each consumer of heat (city district or industry) might have its own heat-producing SMR unit. This idea was present already in the 1970s in the SECURE project of Asea-Atom. Recently several Finnish cities have expressed their interest in SMR-based district heating. Siting near population centers might be justified due to the enhanced safety (lower frequency and smaller activity of potential accidental releases). Also, the alarming situation of climate change calls for carbon-free energy. In Finland, SMRs have even gained some favor of traditionally anti-nuclear political parties.

However, the question of siting options remains to be solved in a rigorous way. At the moment, STUK regulations dictate a 5 km protection zone and 20 km emergency planning zone (EPZ). Smaller requirements would be very beneficial for siting SMRs e.g. at the sites of coal-fired plants.

It is proposed to study the EPZ issue on a sound scientific basis, taking into account the conditions in Finland, not simply following the international developments. Currently it is quite difficult for a small operator to apply for construction and operating licence for an SMR due to the complex licensing procedure that was mainly developed with large NPPs in mind. Difficult issues include spent fuel management and the emergency planning zone. In the US, NRC (independent but possibly influenced by DoE) seems willing to make the situation easier. In Finland, STUK is busy with on-going projects and resources for any preliminary SMR-oriented guidance are probably scarce.

A practical starting point is proposed to be offered for siting and SMR EPZ considerations. This would be realized by selecting SMR types that could practically be viable alternatives for Finnish cities or industry, and studying their core radioactive inventories and release-reducing safety systems to the extent that is possible by public sources. Then, some hypothetical sites for heat production in Finland would be chosen, and their environment studied in detail (population distribution, possible schools / hospitals / other special institutions, traffic connections etc). Calculations of environmental doses to the population would be carried out using selected hypothetical (but justified) fractions of the core inventory possibly released into the atmosphere. Then it can be checked, where STUK dose limits would be exceeded and how the necessary protective measures could be carried out in practice.

In the US, the ORNL (Oak Ridge National Laboratory) made a study of hypothetical SMR sites in 2013: ‘Evaluation of suitability of selected set of DoD military bases and DoE facilities for siting a SMR’ (ORNL/TM-2013/118). A more general study of possible sites was ORNL/TM-2011/157/R1.

An important objective is also to follow closely the international development of the SMR EPZ issue: size of zones suggested by SMR suppliers, on-going work of regulators to tune their guidance for SMR licensing, possible objections by anti-nuclear groups, and the work conducted by IAEA in several SMR-related programmes, like INPRO and the CRP I31029 on SMR emergency planning. However, all the information should be applied for Finnish conditions.

Some more detailed familiarization with suitable SMR designs should be done, as well as selecting some potential candidates for Finland conditions. It is not possible to perform PSA levels 1 and 2 for the candidates (or deterministic analyses by e.g. MELCOR) due to lack of design data, but source term definitions can be done with public information on the power levels, inventories and accident-mitigating safety systems. A review of opportunities among Finnish cities and industry could be performed in order to find some that might practically consider an SMR plant to produce their needed heat. Several potential calculational tools for atmospheric dispersion and dose assessment are readily available.

At present, the possible scaling of SMR emergency planning zones is an open issue. Theoretically, various options for SMR scalable EPZ determination could be available, ranging from simple scaling to laborious analysis through all the PRA levels 1-2-3. As long as the plant is of light water (LWR) type, the reactor core radioactive inventory can be roughly scaled down from large NPPs by thermal power (MWth). When keeping all assumptions of decontamination factors / released fractions, timing of release etc. the same, the dose vs. distance curve will then be scaled down towards smaller doses by the factor of thermal power. However, typical SMR plants have passive safety systems like large water volumes, that could reduce atmospheric release fractions and make large early release less frequent. To account for these, the only completely sound and rigorous way may well be to perform PRA1 to get core damage frequencies (CDF) from various postulated initiating events (PIEs), then PRA2 to get atmospheric releases with their frequencies, and finally PRA3 with site-specific historical weather data to get probability distributions (e.g. CCDF curve; complementary cumulative density function) of offsite doses at a certain time point and distance. The procedure itself is well-known, but complications arise from plant systems of which little experience exists (first-of-a-kind, or FOAK). When level 3 offsite doses have been calculated, they can be compared with generic dose limits for protective measures, which are the same regardless of plant size or type. However, one very basic remaining question is: How safe is safe enough, i.e. should we look at the dose value that can be exceeded in only e.g. 5 % of cases, or choose some other fractile.

Work in EcoSMR

In another task of EcoSMR, the atmospheric dispersion of radioactive material following postulated accidents of two small reactors will be modelled. Releases may be strongly dependent on reactor type, but without detailed information, at least the reactor core radioactive inventory of an LWR type SMR can be assumed by scaling with power level from a large NPP. For NuScale and VTT's own LDR design inventories are available. For more exotic reactor types, even the definition of a severe accident (SA) may not be completely clear. The analysis results will be compared to similar analyses of existing large nuclear power plants, which have been conducted e.g. to check that protective measures would not be needed outside the 20 km emergency planning zone. The amount (activity in Bq of each radiologically significant nuclide) and time distribution of a hypothetical radioactive release from a small reactor will be modelled using conservative assumptions, utilizing models and methodologies usually used for large nuclear power plants. Conservatism is generally not easy to prove, but can probably be claimed justifiably if similar release fractions as for current large GenIII plants are used.

Even for one fixed release source term, it is possible to analyse the effect of changing site specific weather data, depending on data vailability. Then the dose results are not single values, but probability distributions for any combination of receptor point time and location.

The temporal distributions can be different for different nuclide groups: For example, noble gas fission products are expected to escape from damaged fuel and primary circuit rapidly, even when relatively long retention times (i.e. delays between fission power and atmospheric transport begin) can be expected for aerosols particularly in the case of large water volumes (which can also provide a good decontamination factor DF). For these reasons, the nuclide composition of the atmospheric radioactive cloud will generally vary according to the starting time of the currently on-going release stage. An important difference from large plants might be the lower effective release height from an SMR.

The radiation dose to population caused by a postulated severe accident of a small reactor will be analysed at distances corresponding to the possibly relevant emergency planning zone sizes of the examined case. The distances to be carefully analysed should be small, starting from a few hundred meters (or appr. site boundary), for an SMR plant. The effectiveness of countermeasures such as iodine pills, sheltering and evacuation will be modelled. Optionally, consequences such as contamination of food-yielding fields or health effects (minor increase in stochastic ones expected) on population can be evaluated. It is important to note that the traditional conservative assumption of consuming locally produced food only should not be used for a small reactor in an urban setting (the population will consume food from long distances anyway). Any possible contaminated agricultural produce can nowadays be discarded without great difficulty. Generally, some new considerations in determining the representative person of the most highly exposed population group may be needed.

An important aspect of this future task is to model a release from a small reactor significantly closer to population than most of the current large nuclear power plants. This might affect the dispersion, which will now take place in an urban environment, not in relatively open (obstacle-free) countryside. A straightforward conclusion from the urban (dense population) setting is that the same levels of individual doses (Sv) will then inevitably cause more population dose (manSv).

The task will consist of producing some general guidance for SMR EPZ determination, but also practical sample cases to test the methodology and to illustrate the results as close to reality as possible. Being able to provide rigorous and justified determinations of suitable emergency planning zone (EPZ) sizes for SMR plants can prove a valuable service for any company willing to build and operate them. The conditions for siting them near industry and population centers are a central question. Their might be considerable interest for such determinations using site-specific meteorological data and comparing dose results with IAEA generic intervention levels for protecting the public, especially in emerging nuclear energy countries.

For many types of small reactor, including the VTT design of LDR-50 (Low-temperature District heating and Desalination Reactor), there has been discussion about underground siting. That would have clear benefits in excluding some initiating events of accidents and helping to contain possible radioactive releases, but on the other hand, may complicate some conditions, like flood or fire. Most probably however the advantages would outweigh the complications, considering also that probably public confidence in reactor safety would be increased with underground siting of which only access tunnel and some ventilation etc. systems are visible. So it is strongly recommended to study the underground siting option in more detail.

14. Summary

A literature study of SMR emergency zones was performed. The growing international interest in SMR nuclear power plants has included discussion on the required size of their emergency preparedness and response (EPR) distances or zones. Plant providers usually try to justify small zones, whereas the regulators, in many cases, have not made it clear if and with exactly what justification those can be approved during the licensing process. Emergency distances from 1 km to 2 km have been mentioned for e.g. KLT-4S, VBER-300, ABV, SMART and IRIS.

These achievements could be realized taking advantage of the smaller reactor core radioactive inventories and the more advanced safety features of the new plants. However, the possibly smaller EPR distances around new types of nuclear power plants and the possibility to completely do away with EPR arrangements remain controversial.

Ideally, a rigorous analysis of the EPR distances of SMRs should be made based on actual radioactive inventories, modelled DF (leak path decontamination factors) and resulting atmospheric release source terms, as well as a computational assessment of doses and their comparison with international action levels for radiological countermeasures. The main question is how far from an SMR plant the PAZ (precautionary action zone) and UPZ (urgent protective action planning zone) should reach, which is of particular importance if the SMR plants are to be used as a local source of heat for cities and industry.

This literature study task covered emergency planning provided by SMR plant vendors, examples of regulatory policies internationally, and analyses on the topic made by research institutes and consulting firms. Recommendation on the use of level 3 PSA was made based on examples like the Canadian licensing process, and some details of VTT capabilities in that field were given.

This report also contains some information on SMRs in general, international (IAEA) safety standards, and international collaboration where VTT participates.

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Vesa Suolanen vesa.suolanen@vtt.fi Security Level: Email, Account Authentication (None)	VIEWED Using IP Address: 84.248.217.104	Sent: 01 February 2022 | 10:13 Viewed: 01 February 2022 | 14:32

Certified Delivery Events	Status	Timestamp

Electronic Record and Signature Disclosure:
Not Offered via DocuSign

Carbon Copy Events	Status	Timestamp

Witness Events	Signature	Timestamp

Notary Events	Signature	Timestamp

Envelope Summary Events	Status	Timestamps
Envelope Sent	Hashed/Encrypted	01 February 2022 | 10:13
Certified Delivered	Security Checked	01 February 2022 | 14:32
Signing Complete	Security Checked	01 February 2022 | 10:20
Completed	Security Checked	01 February 2022 | 14:32

Payment Events	Status	Timestamps