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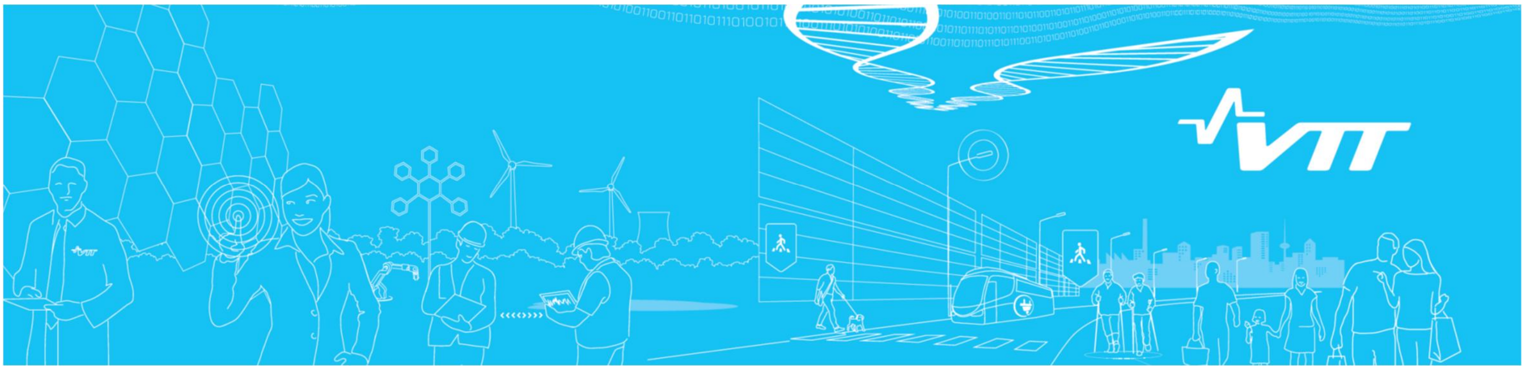

RESEARCH REPORT

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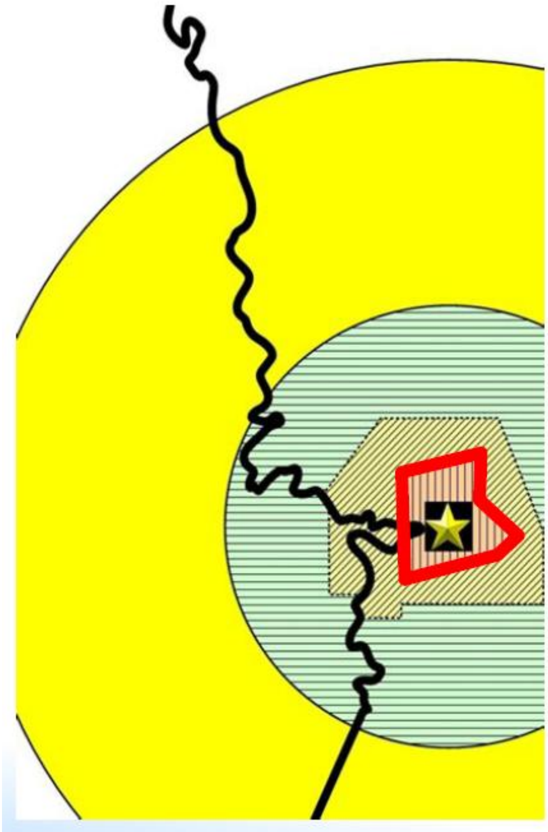

(IAEA)

Off-site radiological consequences from an SMR unit

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Summary

This report considers licensing issues of SMR (small modular reactor) plants through the final level of PSA (probabilistic safety assessment): level 3, or off-site consequences. The report has two main parts: Current international (IAEA) developments in proper sizing of EPR (emergency preparedness and response) zones of SMRs taking into account their enhanced safety features, and a computational assessment by the VTT-made VALMA code (atmospheric dispersion and dose assessment) of the off-site doses caused by a hypothetical radioactive release from a NuScale reactor unit. Because reliable assessment of atmospheric source term is still lacking, many conservative assumptions were used. It is not possible to give unambiguous recommendations on sizes EPZ (emergency preparedness zones) based on the calculations made in this work.

Small modular reactors (SMRs) have been a topic of international discussion in recent years due to many advantages compared to present day large power plants: improved safety features, especially after Fukushima accident in 2011 and reduced capital costs through design simplification/smaller size. Many other applications than electricity, available in SMR concepts, include process heat production, desalination and hydrogen generation. In addition, SMRs are a potential option for developing countries where grid capacity is not sufficient for large NPPs.

For assessing the possibility of SMRs in Finland, the licensing requirements and related safety issues should be studied in more detail. There are also technical challenges that need to be evaluated. This includes particularly passive safety systems and further evaluation of possible off-site consequences.

Confidentiality	Public

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Preface

The main objective of the GENXFIN project is to increase scientific and technological knowledge of new nuclear energy technologies in Finland. This objective is mainly achieved by international collaboration: participation in international working groups and meetings. In 2016-2017 the main focal points were supercritical water reactors (SCWRs) and more generally, the licensing issues of small modular reactors (SMRs). This report considers the latter: the current international (IAEA) developments in determining the sizes of emergency preparedness and response (EPR) zones of SMRs.

Espoo, 31.1.2018

Mikko Ilvonen

Cover page image: IAEA, Technical meeting on next generation reactors and emergency preparedness and response, 13-17 February, 2017

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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 question of licensing and siting should receive more rigorous consideration.

The smaller core 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. So also the environmental consequences of an atmospheric radioactive release from an SMR are expected to be less severe than from typical present LWRs. This fact 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 off-site emergency preparedness. 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.

In this report the VTT-made VALMA code was used to assess the near-range off-site doses caused by release from an Olkiluoto-situated NuScale reactor unit. Real SILAM-calculated NWP-based weather data of year 2012 from the FMI was used. The inventory was acquired from an approximate Serpent calculation, but otherwise many conservative assumptions were used: high burnup for all assemblies, decontamination factor comparable to present large LWR (e.g. EPR), only 4 hours of cooling time between SCRAM and start of release, and all released into the atmosphere within 3 hours.

It would be too daring to try to use 'best estimates' with the resources budgeted for this work in GENXFIN, because there are simply too many unknowns in the NuScale design. Still, the conservative assumptions show that the off-site consequences are clearly lessened when compared with a large LWR.

With the main objective to evaluate and determine EPZs for NuScale, 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 have to be obtained. Atmospheric dispersion conditions affect significantly the doses. Weather data is based on annual data, so a probabilistic approach of doses can be adopted. Exposure pathways include the relevant pathways: external radiation from the plume, inhalation and external radiation from the deposition on the ground. The main desired outcome is to prepare for recommendations, with proper justification, on the emergency planning and response for a NuScale plant. This includes EPZs (emergency planning zones) based on the international safety standards.

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

2. Small Modular Reactor (SMR) basics

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 calculations in this report were done for the NuScale SMR concept. The international development of emergency preparedness and response (EPR) in this report is more general, for any kind of SMRs.

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

- Small power (< 300 MW) and compact architecture
- Passive features and safety systems
- In-factory fabrication of a SMR unit
- 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
- Replace economy of size with economy of modular production in same factory
- Decrease the initial investement needed
- Lower requirement for access to cooling water
- Suitability also for remote regions & low capacity grids
- Possible daily load following (even with intermittent energy sources)
- Possibly reactor module removal at the end of the lifetime
- Electricity generation, district heating, cogeneration, water desalination, high temperature process heat for process industry, hydrogen production
- Output 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]*. LWRs are the most common nuclear designs in the world: there are around 437 reactors in operation and of them 357 are LWRs. Of these LWRs 273 are PWRs, so most experience has been gathered 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.

SMRs of the LWR (light water reactor) 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 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.

The calculations done in this report focus on NuScale, which is an integrated pressurized water reactor, or iPWR. NuScale is aiming to build its first SMR plant in the US by 2023, and believes it could build its first UK plant by the mid-2020s. In January 2017, the NRC reveived the design certification application from NuScale. According to news in January 2018, the NRC then concluded that the NuScale SMR design does not need backup electric power supply of Class 1E. The preliminary safety evaluation report (SER) is expected in April 2018. In some SMR designs, all safety-critical equipment including the reactor and the fuel vessels will be located underground, like NuScale, minimising the need for expensive physical defences.

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, produce serially in a factory (enhancing quality), transport in one piece
- Shorter construction schedules, smaller initial investment
- Lower grid capacity (like in developing countries) enough, smaller backup power need, possibly operate in own grid
- Load following: heat/electricity cogeneration, number of SMR units in production
- Smaller core radioactive inventory, the 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 maintained and power-producing units being located close to each other

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

3. Licensing issues of SMRs

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 pave a way to harmonisation of nuclear licensing internationally *[Söderholm, 2013].* One central question is the usual manner in which STUK controls also 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.

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 (changed 1 January, 2018). 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. 1 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

The 5th level considers the situation where a radioactive release into the off-site 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 off-site 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 1. Defence in depth levels according to WENRA [WENRA, 2013].

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, utilisation 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.

In the USA, the NRC will possibly 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:

- 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, 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 1 below (possible EPZ scalable approach, conditions for different EPZ radii).

Table 1. 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.

4. Short overview of the NuScale SMR concept

Short descriptions of some relevant features of the NuScale design are included here to give a better idea of what is the basis of possible accidents. More complete description can be found in VTT-R-05548-16 [Hillberg et al. 2016]. NuScale Power Modular and Scalable Reactor is an integral pressurized water reactor (iPWR) capable of producing 45 MW of electricity or 160 MW of thermal power. An iPWR system means that the primary cooling system is integrated, i.e. the core, steam generators (SG, helical coil type in NuScale), pressurizer (PRZ), the whole primary circuit coolant and control rod mechanism are located inside the reactor pressure vessel *[Mazzi, 2005]*. Each nuclear plant consists of 1-12 of these modules. Each of the units is housed in its own pressure containment, submerged underwater in a stainless steel lined concrete pool [Fig. 2]. One safety consideration is the possible failure of multiple modules at the same time, from a common cause. The NuScale concept relies on natural circulation both in normal operation and in accident situations. The concept is being developed by NuScale Power LLC. *[aARIS, 2016; Reyes, 2012]* who submitted the design certification application to the NRC in January 2017. The target for the design certification is 2020 and commercial operation is targeted in 2023 *[aARIS, 2016]*.

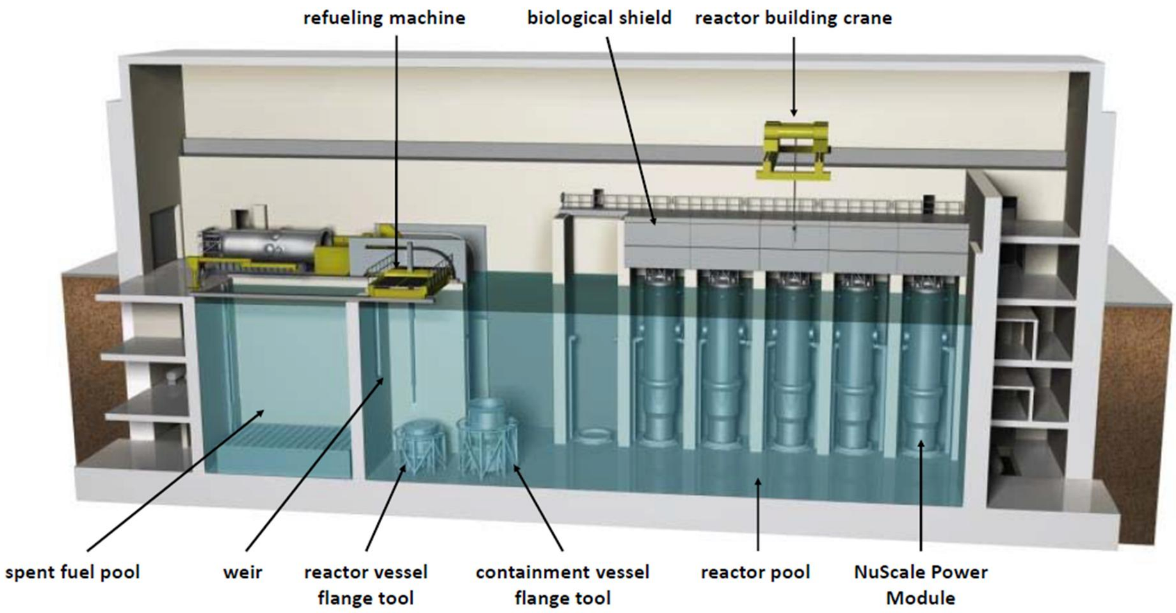

Figure 2. NuScale reactor building cross section [Surina, 2015].

Technology Developer	NuScale Power, LLC
Country of origin	USA
Reactor type	Integral PWR
Electrical capacity (MW(e))	50 (gross)
Thermal capacity (MW(th))	160
Expected Capacity Factor	>95%
Design Life (years)	60
Plant Footprint (m^2)	130000
Coolant/moderator	Light water
Primary circulation	Natural circulation
System pressure	12.8 MPa
Core inlet/exit temperatures (°C)	-- / 302
Main Reactivity Control Mechanism	Control rod drive, boron
RPV Height (m)	17.4
RPV Diameter (m)	2.9
Module Weight (metric ton)	--
Configuration of Reactor Coolant System	Integrated
Modules per Plant	1–12
Target Construction Duration (months)	36
Seismic design	0.5g peak ground acceleration
Predicted core damage frequency (per reactor year)	1E-8 (internal events)
Design Status	Under development

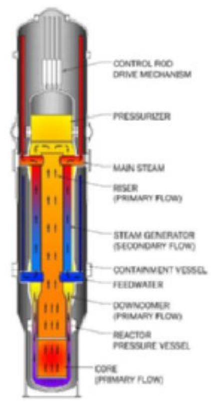

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Figure 3. NuScale plant basic data [Source of figure: IAEA].

Some basic data of the NuScale plant is shown in Fig. 3: Reactor type 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.

Quite interesting results, comparable to what is attempted in this work, 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

4.1 Use of natural circulation in NuScale

A short introduction to natural circulation (NC) is given here. NC is fundamentally different from forced circulation by pumps. For a more profound explanation, see VTT-R-05548-16 [Hillberg et al. 2016], as part of which a literature review was conducted of some prominent, mostly LWR based, SMR designs and problems possibly encountered with natural circulation.

NuScale has no pumps to circulate water through the reactor, but instead relies on the principles of natural circulation. The term ‘natural circulation’ refers to the case in which only naturally occurring forces, like buoyancy / gravity cause fluid to flow, usually in a closed circuit, but in cases also in an open circuit. This is in contrast to forced circulation, in which e.g. electric pumps create a pressure difference that forces the fluid to flow. The most usual case of natural circulation driving force is a system where the fluid is heated at low elevation, it becomes hotter and less dense and starts to rise until being cooled at a higher elevation, from where it will return as colder and denser to the heater part. In NuScale, the primary circulation in normal operation, as well as residual / decay heat removal, is driven by natural circulation. The coolant density difference between the core and the heat exchanger (SG) causes coolant circulation. Also large pools of water, like the NuScale reactor pool with isolation condensers, will develop a 3D natural convection circulating flow when heat is transferred to them, possibly also by a natural circulation loop.

Natural circulation is a result of differing body forces (N/m^3), usually gravity, acting on the coolant at different locations. The difference in force is caused by coolant density difference, which usually results from temperature difference (rising temperature will generally decrease density). It is essential that the heat source (like reactor core) should be located lower than the heat sink (like steam generator). Otherwise the circulation would be halted by stratification of temperatures.

Driving force from gravity is generally weaker than what is achieved by using pumps. (Usual main pump head pressure is in the order of a few bar.) For this reason, flow resistance (friction, turbulence) of the circulation loop must be reduced, e.g. by using larger diameters, less turns, or more hydrodynamically formed parts. Otherwise, coolant mass flux will remain small.

Thus the driving forces are weak and compared to forced circulation reactors, more careful design and analysis tools are needed. The channel power is limited by the mass flux through the core and in order to increase the channel power, the circulation loop resistance is reduced. Small (by necessity) pressure losses may increase the sensitivity to thermohydraulic instabilities and may lead to oscillating conditions especially during the start-up period. In the low-pressure-low-flow conditions, the commonly employed thermohydraulic relationships (various correlations) are not applicable. Furthermore, since the pressure drop is not controlled by a few components but by the friction with all the walls of the whole circulation loop, a detailed computation of flow and heat transfer is unavoidable for reliable simulation. Stability analyses should be carried out also with coupled computations of neutron kinetics and hydrodynamics.

Advantages of natural circulation, as compared with pump-forced flow, can be listed as follows *[Vijayan & Nayak, 2010]*:

- Construction and maintenance is simpler, as there are no moving parts, leading to lower costs. In addition to the absence of moving parts, the geometry of the flow circuit is usually simpler, because there are less pipe bends, elbows, etc. - often of necessity (to reduce the resistance).
- Usually a substantial part of nuclear power plant accident scenarios result from pump events. All of these are readily eliminated when there are no pumps.
- For electric pumps to work under all conditions, emergency diesel generators or batteries are needed. For natural circulation, these active power supplies are not needed.
- Without pumps, there are fewer connections and so fewer potential leak sites in the system. There is also less connecting piping, the extreme case being the integrated

pressure vessel (like NuScale) containing steam generators. All of this results in fewer possible accident scenarios.

- Natural circulation may achieve better, uniform flow distribution, particularly important for the reactor core. In a typical PWR, a steam line break results in drop of secondary pressure and rapid cooling of the primary loop going through the affected SG. In normal operation, a flow distribution device may be used to direct core inlet flow. In natural circulation, the thermal driving head is greater for high-power channels. Furthermore, the driving force of possible colder water is readily decreased.
- Natural circulation flow, particularly two-phase flow, will increase with heating power, whereas forced two-phase circulation meets more resistance with power, as more bubbles are generated.
- In natural circulation, low flow velocities and reducing friction lead the design to large volumes and low power densities, which is inherently safer than higher power density and less coolant, because there will be large thermal inertia (slow thermal response) in the system as more time will be needed to heat the coolant to a certain temperature.

Possible problems in natural circulation

A joke about passive systems says that they may be ‘too passive’, i.e. not start to work at all when needed. Ultimate proof could be to check in the real plant, before loading with nuclear fuel, as many of the passive systems as possible for proper functioning when the starting condition is triggered. For natural circulation, many kinds of physical phenomena could prevent proper function: thermal stratification, a steam void blocking circulation, loop seal (blockage of primary coolant loop with filled water), or various manometer-type effects.

Vijayan & Nayak (2010) list several kinds of challenges that may compromise the proper functioning of natural circulation flow: low driving force in comparison with pumps, the resulting need to design for low pressure losses, low mass flux, various instability effects, problems specific to LPLF (low pressure low flow) regime, difficult start-up and operating procedures, and possibly low value of CHF (critical heat flux).

In addition to instabilities and pressure losses, some further problems of natural circulation are related to control and operating procedures, CHF margins, and difficulties in simulation models:

- Low mass fluxes mean lower maximum channel power. To produce enough total reactor power, it is then necessary to use larger core volume, which in turn may bring zonal control & stability problems.
- It is relatively difficult to design start-up and shutdown procedures to avoid instabilities. The required procedures may be complicated. Questions include what is the optimal pressure to initialize boiling, should an external pressurizer (for cold start) be used, and should inlet subcooling be controlled.
- At the start-up of natural circulation plant, the conditions are low pressure, low temperature and no flow (LPLF regime). Powering up from this to nominal operation may involve passing through an unstable zone, risking premature CHF. For a CHF correlation for NC / SMR, see Luitjens (2016).
- Natural circulation is harder to simulate properly than its pump-forced counterpart. A clear reason for this is the sensitivity to small changes in driving force and friction. Another problem is the poor availability of validated TH correlations particularly in LPLF conditions. This applies to system codes and CFD, but the problem is generally worse for CFD, because correlations based on local values would be needed, and their availability is worse than for correlations based on bulk quantities.
- A natural circulation plant will have to comply with both thermal and stability margin requirements. Usually the lower (Type I, low-quality flow, occurring with a chimney)

instability threshold < CHF value < upper (Type II, high-quality flow, most common) instability threshold.

- Design may be complicated by the fact that factors such as inlet subcooling & bottom peaked power have opposite effects on CHF and stability. For example, increased inlet subcooling makes the stable region narrower.

4.2 Other passive safety systems in the NuScale design

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). They are briefly listed here for easy reference, because the probability and magnitude of atmospheric radioactive releases is basically dependent on their abilities. The listing is based on VTT-R-05548-16 [Hillberg et al. 2016], in which the reader may find more information.

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.

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 NuScale concept includes multiple inherent / passive safety features *[[a]ARIS, 2016; [b]Reyes 2012], [Liu and Fan, 2012], [Ingersoll, 2011]*:

- During normal power operation the containment has an isolating vacuum:
    - Reduced heat loss from the reactor vessel (RPV)
    - No RPV surface insulation needed; no sump screen blockage
    - Very little non-condensable (NC) gases in vacuum leads to enhanced condensation rates, if safety valves vent steam into this space.
    - No combustible hydrogen-air mixture; no passive autocatalytic recombiners (PAR) needed
- Small diameter containment with design pressure of 41 bar (cf. system pressure 128 bar), much more than typical current containment structures
- Small core, large water inventory
    - RPV water volume to thermal power ratio is four times larger than that of a conventional PWR. Increased thermal inertia slows down temperature increases and the operators have more response time.
    - Inherently more heat transfer area per thermal power. If the diameter of an SMR core is 1/n of a large reactor, the relative surface area of RPV per unit power is appr. n times of a large reactor.
- Core cooled entirely by natural circulation, eliminating pumps, pipes and valves, and their possible failures. Furthermore, the RPV height-to-diameter ratio is larger than in a typical large reactor (more components vertically inside), which increases gravity-driven natural convection circulation capability.
- Module submersion in a water pool has many safety advantages:

  - ECCS (emergency core cooling system) uses the pool
  - DHRS (decay heat removal system) uses the pool
  - Pool dampens seismic events
  - Additional fission product barrier
  - Radiation shield
  - Pool holds all the water needed for cooling the reactors already in place before any event.
- The integral configuration eliminates the possibility of a traditional large break loss of coolant accident (LBLOCA).
- The water pool containing the reactor modules is inside a steel-lined, pre-stressed, post-tensioned concrete containment capable of withstanding an aircraft impact.
- Under-ground construction, which significantly hardens the plant against external impacts such as aircraft or natural disasters. Below-grade construction of the reactor and containment vessels also has additional seismic resistance and helps to reduce the number of paths for fission product release in the event of an accident.
- Intrinsically smaller radionuclide inventory, with additional barriers to fission product release, leading to dramatically smaller accident source term
- Smaller ratio of volume to surface area leads to better decontamination factor (DF) of fission products.

4.3 Severe accident in a NuScale plant

The possibility of a severe accident (one with core melting at least partially), even though unlikely, has to be taken into account in the design of 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 SAM in NuScale and also with some description of severe accident phenomena in general. It is mainly based on VTT-R-05548-16 [Hillberg et al. 2016], which the reader can refer to for more information. At VTT, severe accident simulations are mainly conducted by using the integral codes ASTEC and MELCOR.

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

The NuScale management methods of the severe accident phenomena, with higher probability of a significant role, which might threaten the containment integrity are presented in Table 2.

Table 2. Severe accident management in NuScale. Containment phenomena.

	Core melt management	Hydrogen management	Fission product transport	Containment heat removal
NuScale	In-vessel melt retention (IVMR)	No combustible mixture of hydrogen and oxygen inside containment (high vacuum)	Additional barriers: pressure vessel housing the RPV, module submerged in water (steel-lined container), biological shielding for each module, scrubbing in reactor pool	Passive heat exchangers (single-failure proof) 3 d, reactor/containment circulation 30 d, air cooling infinite

The main basic differences of NuScale from large reactors, affecting the probability, progression and possible off-site consequences of severe accidents, can be listed as follows:

- Integrated design (iPWR) of the primary circuit components in the RPV. Because the steam generators, pressurizer and the control rod mechanism are all inside the RPV, there are no large penetrations and pipelines. Large-break LOCA accidents are inherently prevented, because failures of those structures are not possible. These features may also limit the scope of small (SB) and medium-break LOCAs *[IAEA, 2009]*. The integrated control rod mechanism prevents reactivity accidents by control rod ejection: There is no pressure difference that would cause such an event.

- The NuScale design used in this report has only 160 MW of thermal power. Then the decay heat is lower compared to ‘large’ plants, roughly by the same ratio as the nominal power of the plants. The low power, passive safety systems (which operate without external power) and relatively large coolant inventory assist in the safe termination of accidents.

- Smaller reactor core radioactive inventory of fission products etc. Starting from this premise, the residual decay heat, and also the possible atmospheric release source term is smaller. The environmental consequences of a radioactive release from an SMR may thus be expected to be less severe than those from larger reactors.

Residual heat removal from core and containment

NuScale is relatively well documented in public literature. The concept relies on independent passive safety systems. However, it seems that in some cases opening of valves etc. may be needed. Also primary coolant circulation under normal operation relies on natural circulation. Each module includes two redundant passive safety systems to provide pathways *[Liu and Fan 2013]* for decay heat to reach the containment pool: decay heat removal system (DHRS), and containment heat removal system (CHRS). These systems do not require external power for actuation. The NuScale layers of barriers between fuel and environment are:

- Fuel pellet and cladding
- Reactor pressure vessel (RPV)
- Containment (shaped as RPV, but somewhat larger)
- Water in reactor pool
- Stainless steel lined concrete reactor pool
- Biological shield covers each reactor
- Reactor building

The DHRS (decay heat removal system, leftmost third of Fig. 4, 'water cooling') can transfer core decay heat from either of the two SGs to isolation condensers (IC) immersed in the reactor pool. The system is a closed-loop, two-phase natural circulation cooling system with two redundant trains attaching to each of the SG loops. The DHRS is capable of decay heat removal for a minimum of 3 days without pumps or power *[[a]ARIS, 2016; [a]Reyes, 2012].*

The ECCS (emergency core cooling system) consists of the reactor vent valves (RVVs) located on the RPV head and the reactor recirculation valves, located on the sides of the RPV. ECCS operates by opening the vent valves which vent primary system steam into the containment to be condensed on the containment's inner surface. The condensate collects in the lower region of the containment vessel. When the liquid level in the containment rises above the top of the recirculation valves, valves are opened to provide natural circulation path from the lower containment through the core and out the RVVs. The system works in conjunction with Containment Heat Removal System (CHRS) which appears to mean the passive convection and conduction heat transfer of the outer surface of the containment vessel *[[a]ARIS, 2016; [a]Reyes 2012].* 'Boiling' and 'air cooling' in Fig. 4 mark the two phases of ECCS.

According to the manufacturer, the single-failure proof passive heat exchangers in the containment vessel provide cooling for 3 d (containment pool still filled with water). After that convective circulation and boiling in the reactor pool suffices for heat removal for 30 d. For dry reactor pool, air cooling is sufficient to remove the remaining decay heat for an infinite period *[[b]Reyes, 2012]*.

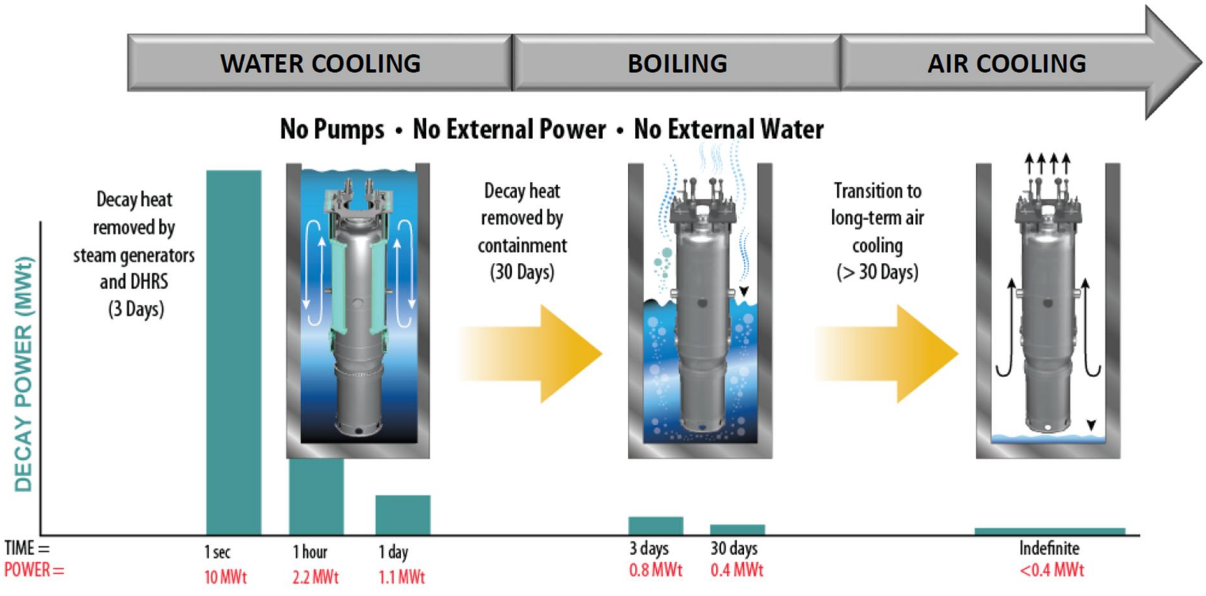

Figure 4. Residual heat removal in the NuScale design. Plant response to loss of all power situation [Surina, 2015].

5. The ideal procedure to determine EPZ sizes

The budgeted resources in GENXFIN for mechanistic determination of NuScale emergency planning zone (EPZ) sizes were quite small and did not allow for a complete, ideal work procedure. However, the necessary phases of such a procedure are outlined here for future reference.

With the main objective to evaluate and determine EPZs for NuScale, 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 have to 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 condition or the measured annual data. In the latter case a probabilistic approach of doses 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 a NuScale 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 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 off-site 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 for research work on EPZ size are:

- EPZ sizes given by SMR designers in general
- Design of the NuScale reactor
- Site area and EPZ size of NuScale by design
- IAEA Requirements and Guides for EPR (emergency preparedness & response)
- Potential codes for determining the release to atmosphere
- Potential codes for off-site dose assessment

Core radioactive inventory of NuScale should be calculated or acquired from designer.

List and description of NuScale postulated DBA / DEC accidents should be available, either from designer or by VTT assessment.

Possibility of a severe accident (with core melt) 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.

Estimates of the atmospheric release source term:

- Direct information from designer
- Expert judgement 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 probabilities

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. (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.)

Performing the complete plant-specific PSA/PRA, including also deterministic investigations of phenomena, would have the following phases:

- Level 1 (core damage frequencies):
  - Transient scenarios leading to core damage
- Level 2 (atmospheric releases with their frequencies):
  - Inventory, release from fuel, release through containment barriers
- Level 3: Off-site doses with their frequencies:
  - Use real site-specific weather data of several years

Accounting for collocation (multiple units) has been studied at VTT by Ilkka Karanta in the PRAMEA project.

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.

Possibly the EPR criteria should be based on a maximum credible accident: Worst case (deterministic) radioactive source term from containment into the atmosphere should be determined. For all possible source terms to be used, the definition should ideally consist of nuclide-specific released activities, temporal distribution of release rates (Bq/s) and their initial distribution along the vertical (height) axis.

Consideration of site-specific conditions (weather, surrounding environment, population centers etc.) should be included if a possible site for the plant is known.

Selecting and acquiring representative weather data:

- Existing site weather mast data
  - 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.)
- Other generic source of real weather data ?

Off-site dispersion and dose assessment (public doses) calculations should be done with a selected code or codes:

- 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

The probabilistic 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 results. 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 weather situations.
- Distances from plant representative of possible (to be determined) extent of the EPZ

The first interesting comparison would be to compare the off-site doses with a generic large power reactor (size of e.g. Olkiluoto-3).

The predicted dose levels should be extensively compared with STUK / IAEA criteria for protective radiological countermeasures, like sheltering indoors, iodine pills or evacuation. There are two essential consderations: is the criterion exceeded, and if it is, will it be practically possible to perform the countermeasure action?

Only then is it possible to give justified recommendations of EPZ size, based on

- expectedly needed protective measures, or
- possible appropriate scaling down from large reactor EPZ.

A complete protection strategy should include the following definitions:

- EAL: emergency action levels (threshold for a plant condition to decide the emergency class
- OIL: operational intervention 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

Consideration of all hazards, together with all their consequences, is important. Also the dose-averting countermeasures have radiological and non-radiological (societal, economic) consequences.

6. NuScale core radioactive inventory

The reactor core radioactive inventory of NuScale used for the VALMA off-site dose assessment in this work was calculated by Riku Tuominen of VTT using the Serpent code. In the past similar tasks were usually done with the Origen code. Serpent had already been used in the previous 3SMR work of 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 Monte Carlo code and thermal-hydraulics with COSY (Component/System-scale) thermal-hydraulics (TH) tool. Both of the codes are developed at VTT. A very short description is included here for easy reference. More about the 3SMR work can be found in VTT-R-05548-16 [Hillberg et al. 2016].

The data available to public was very limited and detailed core design specifications for NuScale wasn't found online. Most of the available data was in the form of presentation slides. However, based on the information found a mock-up SMR core for the test calculation was created based on the NuScale design.

Radial layout of the Serpent model is presented in Fig. 5. Some of the data required for the modelling was acquired from NuScale documents *[NuScale Power, 2014; Linik, 2015]* but since no detailed core specifications were available, many expert-judged guesses were involved in the modelling.

Compared with typical current PWRs the most obvious difference is the size of the mock-up SMR core. There are only 37 fuel assemblies and the total thermal power is 160 MW. Active fuel height is 200 cm. For comparison, those values for the EPR unit (Olkiluoto 3) *[TVO, 2016]* being constructed in Finland are 241 fuel assemblies, thermal power of 4300 MW and active fuel height of 420 cm. The fuel assemblies in NuScale are standard 17x17 square assemblies used in western PWRs. The cladding is standard Zircaloy 4. The fuel is UO_2 with U-235 enrichment of approximately 2.0 percent. For simplicity the enrichment was the same in all of the assemblies and fuel rods. This enrichment was chosen to achieve an approximately critical core when neutron absorbing boron was added to coolant.

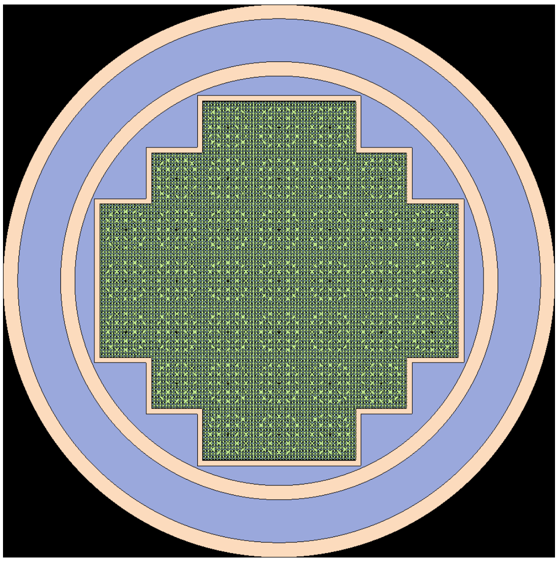

Figure 5. Radial layout of the Serpent model. The 37 square fuel assemblies in the center of the core are surrounded by core baffle.The smaller ring is the core barrel and the larger ring is the pressure vessel. Source of Figure: VTT-R-05548-16 [Hillberg et al. 2016]

Some information on the Serpent-calculated NuScale core radioactive inventory:

- Approximation, based on previous steady state inputs
- NuScale resembles usual LWR reactors > no big differences expected
- Results for 1380 nuclides & 46 burn-up time points (to 50 MWd/kgU)
- No cooling time (can be calculated by VALMA, only 4 h will be used)
- VALMA uses appr. 100 important nuclides
- Some 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 aver.)
  - NuScale 50 MWd I-131 (8 d) 1.7e17 Bq (1/25)

7. The VALMA model, weather data and other input

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.

The weather data used

The weather data was provided by the FMI (Finnish Meteorological Institute). The data consists of the SILAM-calculated air parcel trajectories (no mass) of the year 2012 based on the numerical weather predictions (NWP) of ECMWF (The European Centre for Medium-Range Weather Forecasts). The data covers the grid area of 1000 x 1200 km (56.8137...65.6583N, 10.7129...32.1711E). The calculation resolution of the ECMWF data was 16 km.

There are 20 trajectories in every 12 minutes resulting in the 100 trajectories in one hour. The total number of the trajectories is 878400 (2012 was leap year). Each trajectory is followed for 96 hours if not leaving the calculation area.

The release point is Olkiluoto and the release height of the trajectories was 0 - 200 m. For the current calculations the trajectories starting between the altitudes of 80 and 120 m were sampled for the calculations. This corresponds to an assumed cloud rise due to its initial heat content. For future work, also other release heights should be used. Furthermore, higher-resolution NWP data should be acquired for the near-range calculations. The data set used here was originally used in the CASA project for dispersion up to 300 km distance. The SILAM-calculated dispersion data set can be summarized as follows:

- Calculated by Julius Vira (FMI), October 2015
- 1 Jan … 31 Dec 2012
- For Olkiluoto NPP site
- Air parcel trajectories (i.e. massless particles)

- No gravitational settling, even for aerosol form particles
- ECMWF (European Centre for Medium-range Weather Forecasts) numerical weather prediction model
- Horizontal resolution of NWP data was 16 km x 16 km
- 20 trajectories every 12 min (= 100 / hour), appr. 9 GB total
- A total of 878400 trajectories, followed for max. 96 h each
- Release height 0…200 m
- In VALMA, it was specified: 4 h cooling time, 3 h release duration

Source term and grid specifications

Because no modelling-based (MELCOR etc.) or published information about NuScale accidental atmospheric releases was available, the source term was set by using certain expert judgement, which is briefly explained here. In a previous project (CASA), a severe accident release ('CASA1' source term) from Olkiluoto-3 EPR, defined by STUK, has been used. It contains 100 TBq of Cs-137, among many other nuclides, released into the atmosphere. 100 TBq is the limit of severe accident release set in Finnish regulations. Using knowledge of EPR core inventory, the release fractions of the CASA1 source term were calculated. As Olkiluoto-3 EPR has been successfully licensed in Finland, we can assume that its release barriers are sufficient to decontaminate the release with the factor DF that can be calculated, nuclide by nuclide, by fractions of the EPR inventory contained in the CASA1 source term.

Then, making a conservative assumption, the same release fractions were used for the NuScale inventory. In reality, the fractions should be much smaller, due to the enhanced safety systems and release barriers. The resulting release fractions, by element groups, are shown in Table 3 below. All the fractions could not be calculated in the way explained, and then the fraction of the next more volatile group was used (another conservative assumption). It is assumed in VALMA that only 4 hours of decay cooling took place before the release into the atmosphere starts, and the release duration is only 3 h. Both are very conservative assumptions, probably leading to higher doses than would be reality with the NuScale reactor unit.

Table 3. Released fractions (of NuScale core radioactive inventory) used in the VALMA source term. The final results in this work were calculated from this source term.

```
1    0.0114  'fr_noble_gases'
2    0.0     'fr_organic_iodine'
3    2.4e-4  'fr_iodine'
4    1.8e-4  'fr_alkali_metals'
5    2e-4    'fr_metalloids'
6    2e-4    'fr_alkaline_earth'
7    2e-4    'fr_transition_metals_1'
8    2e-4    'fr_transition_metals_2'
9    2e-4    'fr_lanthanides_1'
10   2e-4    'fr_actinides'
11   2e-4    'fr_transition_metals_3'
12   2e-4    'fr_lanthanides_2'
13   2e-4    'fr_misc_1'
14   2e-4    'fr_transition_metals_4'
15   2e-4    'fr_misc_2'
```

Doses were calculated for a set of receptor points, located at 3 degree intervals in the lateral (angle) polar coordinates, at distances of 1, 2, 3, 5, 8 and 12 km from the release source. The time points of dose integration, to be used for comparison with countermeasure limits, were 3.5 h, 1 d, 2 d, 1 week, 1 month and 1 year.

Some sample calculation cases from VALMA

Before the final VALMA computations, some test cases were run with a very big source term, releasing e.g. the whole noble gas inventory and 70 % of iodine to the atmosphere (see Table 4 of release fractions of element groups below).

Table 4. Release fractions in the test cases with a big release. This source term was only used for the graphical (map) examples of how VALMA follows the radioactive cloud.

```
1    1.0      'fr_noble_gases'
2    0.0      'fr_organic_iodine'
3    0.7      'fr_iodine'
4    0.7      'fr_alkali_metals'
5    0.7      'fr_metalloids'
6    0.1      'fr_alkaline_earth'
7    0.4      'fr_transition_metals_1'
8    0.005    'fr_transition_metals_2'
9    0.005    'fr_lanthanides_1'
10   0.005    'fr_actinides'
11   0.005    'fr_transition_metals_3'
12   0.005    'fr_lanthanides_2'
13   0.005    'fr_misc_1'
14   0.005    'fr_transition_metals_4'
15   0.005    'fr_misc_2'
```

The release fractions of the big source term were partly based on the WASH-1400 report (NRC 1975) and NUREG-0771 (Pasedag et al. 1981); see Table 5 below:

Table 5. 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		$6.4 \cdot 10^{-3}$	$6,9 \cdot 10^{-3}$	$5,6 \cdot 10^{-3}$	$6,9 \cdot 10^{-5}$		

Because graphical presentations were not generated in the final runs of the whole year 2012, some pictures from the large release test cases are included here, to give an idea how a single dispersion case in VALMA may look like. In Figures 6-9, the VALMA map area is appr. 40 km x 40 km, and each colored square is appr. 800 m x 800 m. Olkiluoto is in the middle of the map, and the city of Rauma is marked with grey color:

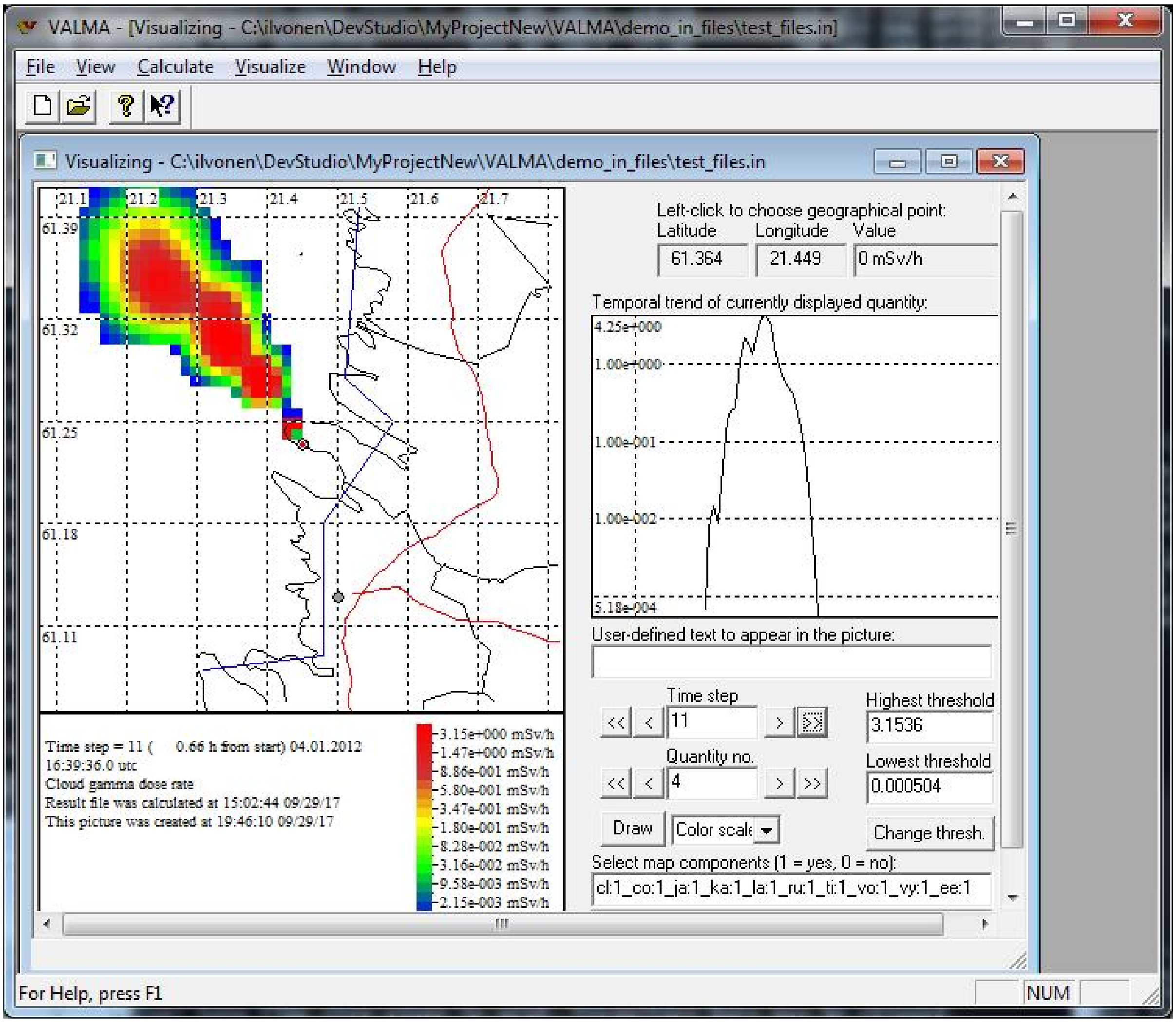

Figure 6. VALMA results for one single dispersion case. Cloudshine dose rate at 40 min after start of release. The big release (Table 4) was used for this example.

Figures 6-8 show the cloudshine gamma dose rate, an instantaneous quantity, in mSv/h. They show clearly how the weather parameters may change in SILAM & VALMA. The radioactive cloud starts spreading to northwest, but then the direction of spread changes to northeast (i.e. southwestern wind), and the already dispersed part of the cloud starts moving in the new northeastern direction. The latter parts, still being released, move to northeast right from the start. The cloud dose rate pictures show the cloud at 40 min, 1 h 52 min, and 3 h 4 min after the start of the 3 h release. The last picture from VALMA (Fig. 9) shows the resulting total dose rate from cloudshine and groundshine. It is the ‘fallout footprint’ of the cloud.

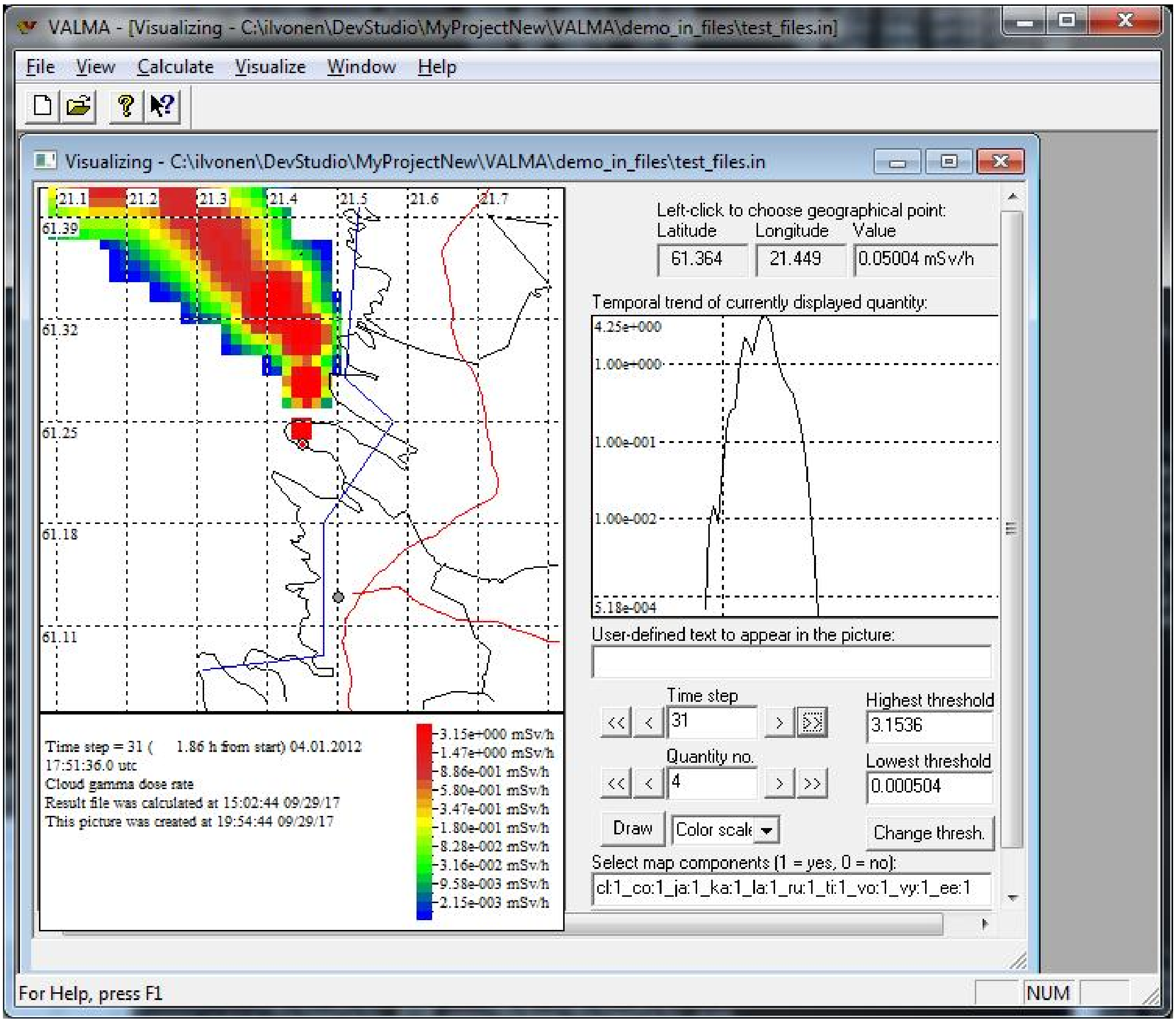

Figure 7. VALMA results for one single dispersion case. Cloudshine dose rate at 1 h 52 min after start of release. The big release (Table 4) was used for this example.

The curve in VALMA visualization is the temporal trend of the currently displayed quantity, located at (61.364 N, 21.449 E) in the example. It is clearly seen how the cloudshine dose rate rises, achieves its maximum value, and then falls again back to zero during passage of the radioactive cloud. On the other hand, the fallout gamma dose rate is very different: After rising and achieving its maximum value, it starts to decrease only quite slowly, because of the radioactive deposition left on ground surface. This behaviour is seen clearly in the temporal trend of Fig. 9, which is the sum of cloud and fallout gamma dose rates.

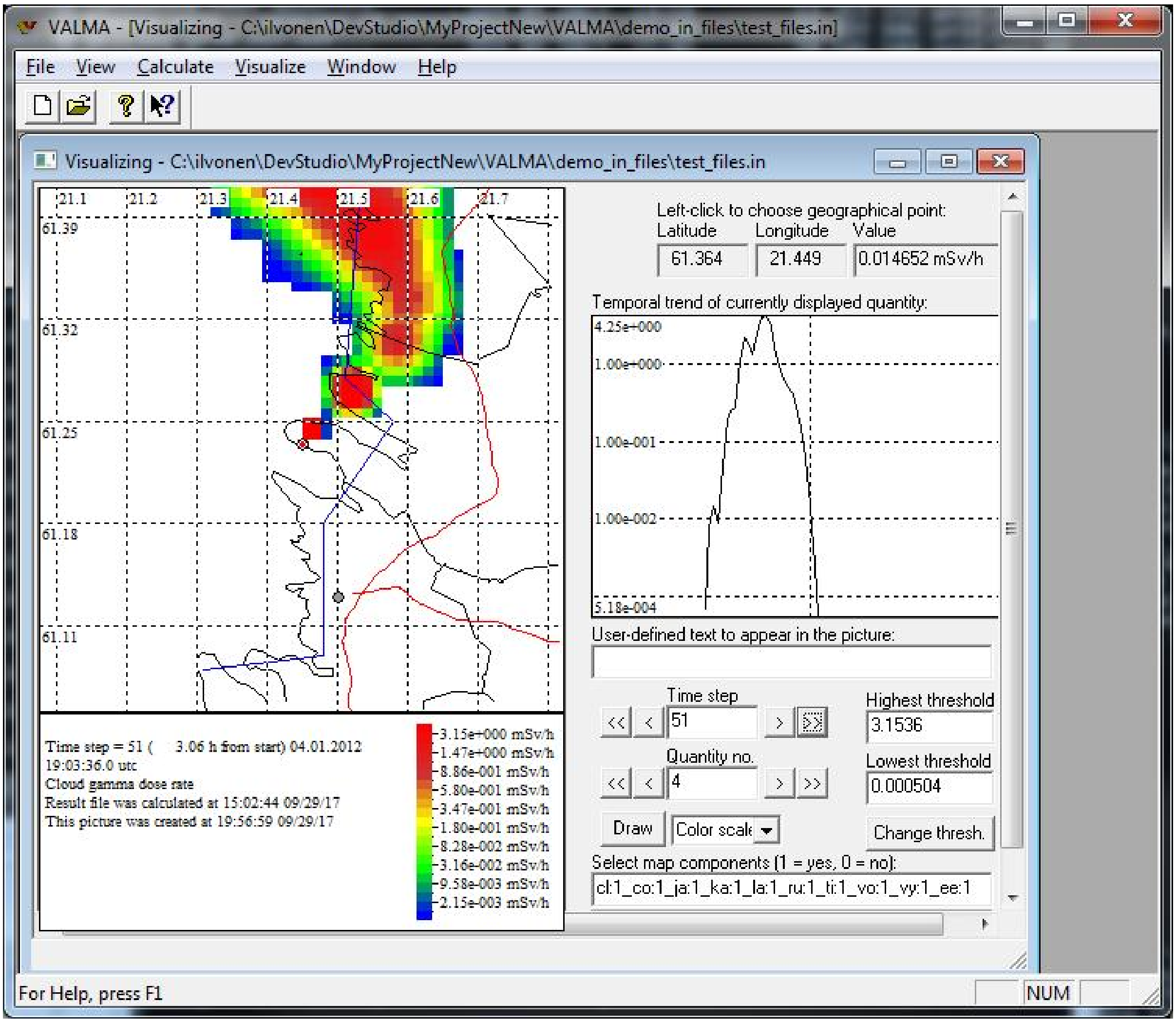

Figure 8. VALMA results for one single dispersion case. Cloudshine dose rate at 3 h 4 min after start of release. The big release (Table 4) was used for this example.

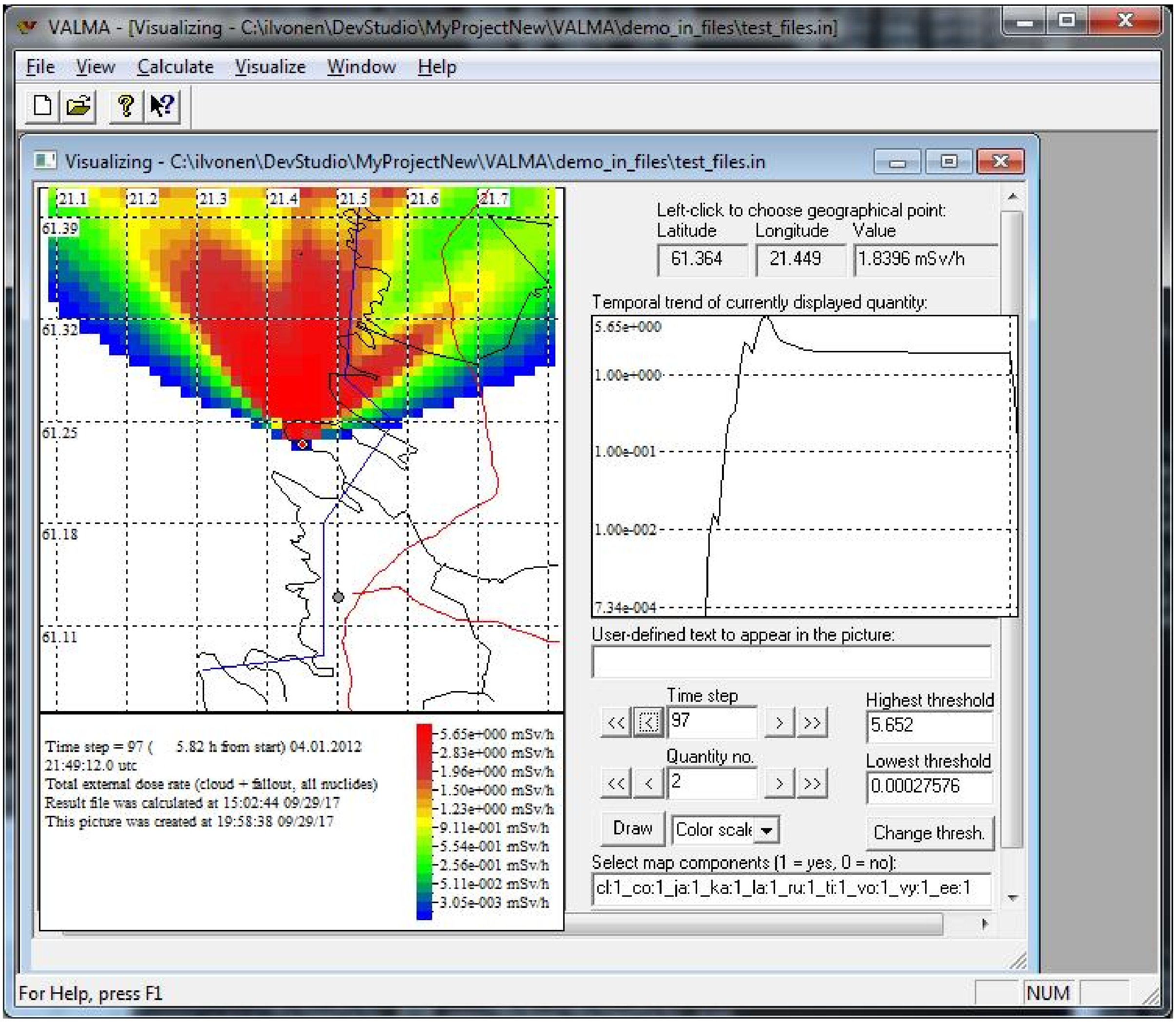

Figure 9. VALMA results for one single dispersion case. Total external dose rate (cloudshine + groundshine) at 5 h 49 min after start of release. The big release (Table 4) was used for this example.

8. Results: Complementary cumulative density functions

From all the 366 x 24 = 8784 dispersion cases calculated by VALMA, a large number of differently defined ccdf curves of the doses were generated. The ccdf means ‘complemntary cumulative density function’, and has traditionally been used to present dose assessment results in a probabilistic sense. The value of the curve is in its easy interpretation: By looking up any value of dose from the x (horizontal) axis, one can read from the y (vertical) axis the probability of exceeding that dose, when all the dispersion cases are considered. The various definitions in this case lead to 540 different curves, of which only a few are shown in Figures 10-12 below:

- Time point (6): 3.5 h, 1 d, 2 d, 1 week, 1 month and 1 year
- Distance (6): 1, 2, 3, 5, 8 or 12 km
- Dose pathway (5): cloud, fallout, inhalation, direct sum, shielded (factor 0.5 for groundshine) sum
- Statistic (3) used to represent the dose at the distance (mean, median, or maximum)

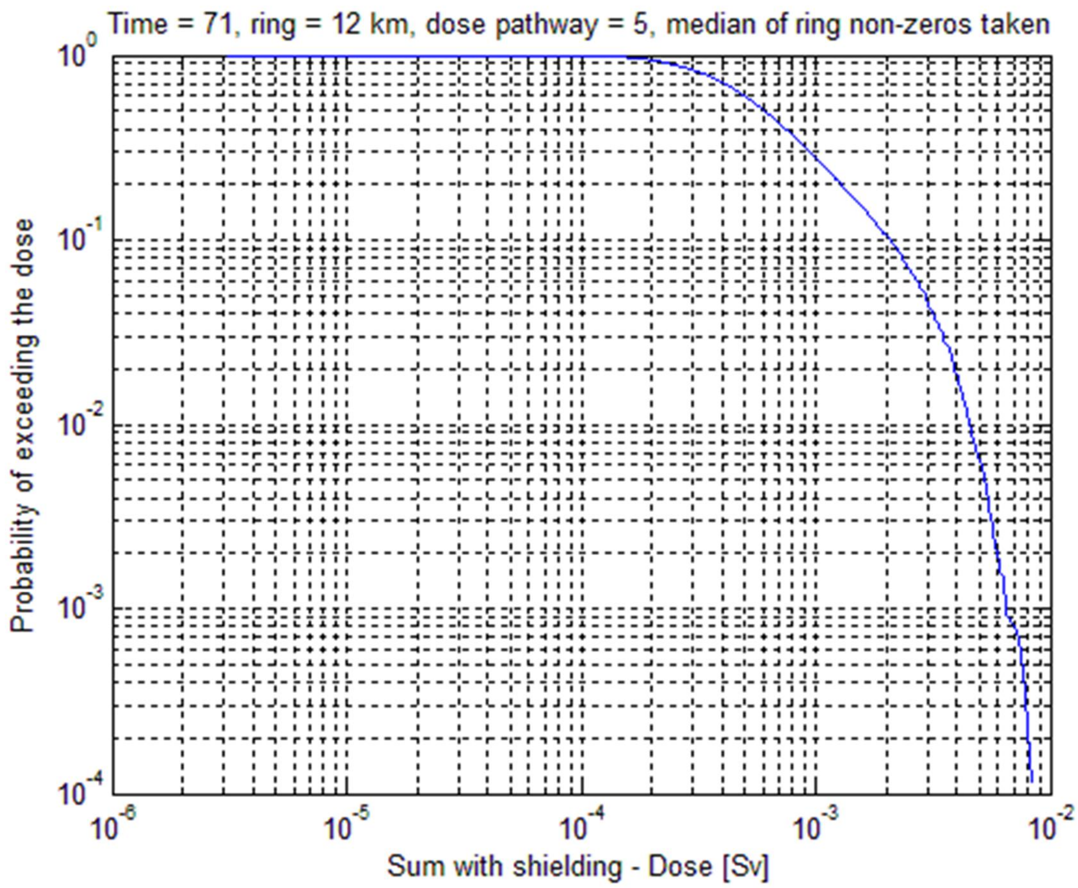

Figure 10. The ccdf curve for time point 24 h, distance 12 km from source, sum of cloud + fallout shielded + inhalation dose. Median of all calculated values at the given distance was taken to represent the dose at the distance. It can be seen that e.g. 1 mSv will be exceeded with 30 % probability among all the dispersion cases in year 2012 weather data.

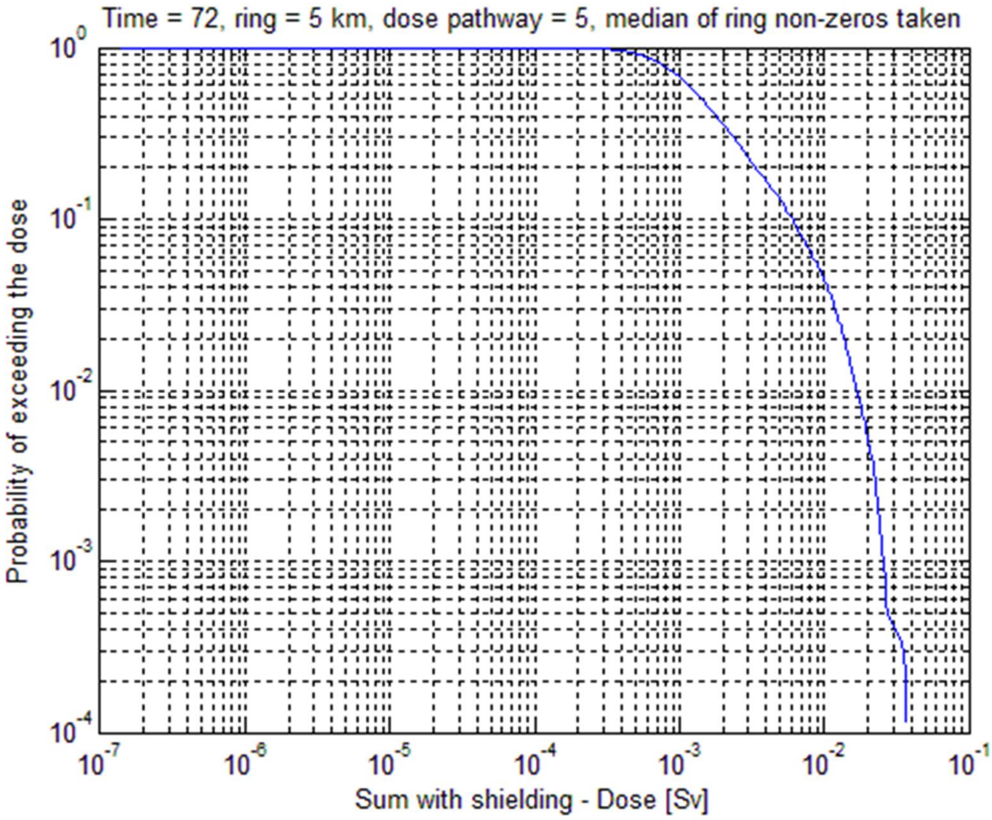

Figure 11. The ccdf curve for time point 48 h, distance 5 km from source, sum of cloud + fallout shielded + inhalation dose. Median of all calculated values at the given distance was taken to represent the dose at the distance. It can be seen that e.g. 10 mSv will be exceeded with 4.5 % probability among all the dispersion cases in year 2012 weather data.

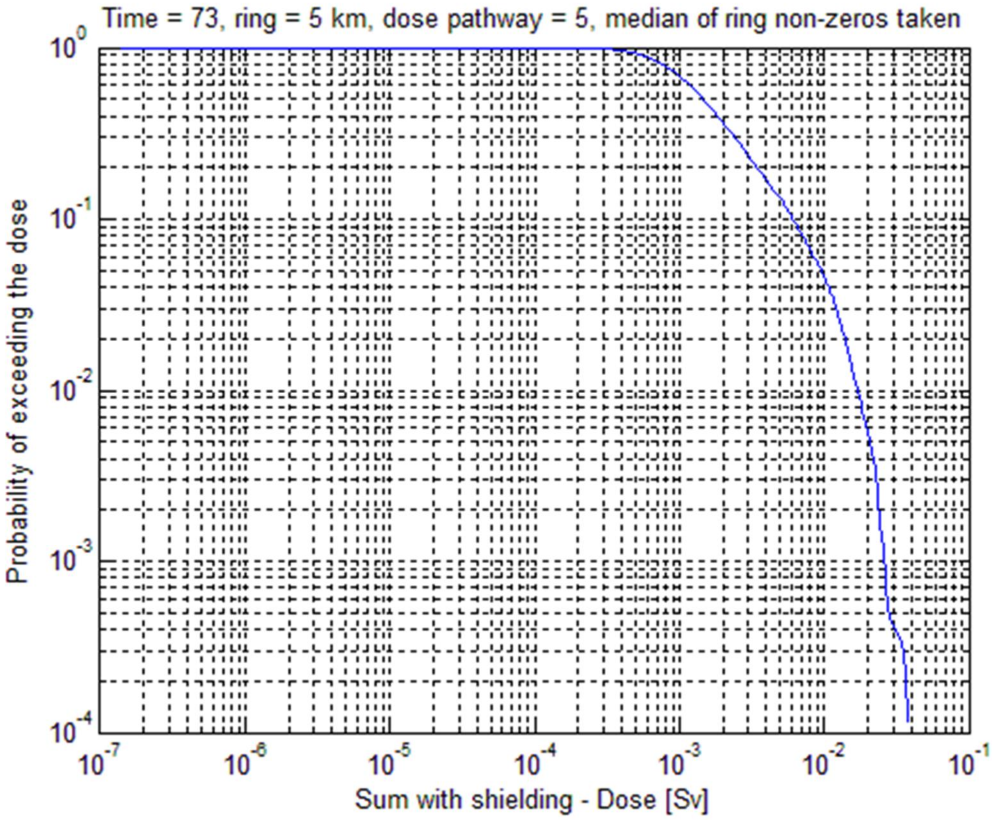

Figure 12. The ccdf curve for time point 1 week, distance 5 km from source, sum of cloud + fallout shielded + inhalation dose. Median of all calculated values at the given distance was taken to represent the dose at the distance. It can be seen that e.g. 20 mSv will be exceeded with 0.5 % probability among all the dispersion cases in year 2012 weather data.

For information of other ccdf results, see the Table in the Appendix. It is not practically possible to understand the ‘big picture’ of doses, their decrease with time and distance, and their probabilities of occurrence by looking at 540 ccdf curves. To crystallize the information, 4 short tables (Tables 6-9) are provided below. in each of them, the sum with shielding was chosen as the dose pathway, and median or maximum appearing at the given distance was chosen to represent the dose at distance. As the fractile, either 95 % or 99.5 was chosen. For example, the latter means that only 0.5 % of all the 8784 dispersion cases had a larger dose. Each table has the time point as row and distance as column.

Common sense suggests that dose should always increase with time and usually decrease with distance. (Note: with real weather in VALMA, it is possible that in certain dispersion cases, the dose increases with distance, usually because of rain occurring there.) Similarly, one expects the ‘maximum’ table to always have greater doses than the ‘median’ table, and also the ’99.5 % fractile’ table to have greater doses than the 95 % one. When checking for these assertions, it can be noticed that the dose did not strictly always increase with time (e.g. 1 km distance in the median & 95 % table). These deviations from expected seem to be of purely numerical origin. (Note that in VALMA, almost all operations of calculation proceed in a numerical manner, and so may depend on e.g. the temporal and spatial grids.)

A first and easy comparison can be made with IAEA GS-G-2.1 (Arrangements for preparedness for a nuclear or radiological emergency) which recommends, as a 'rule of thumb', for 160 MWth:

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

It is very difficult to give any unambiguous recommendations on the EPZ size based on Tables 6-9, but some considerations are given here (cf. Chapter 'IAEA / STUK criteria for protective countermeasures'):

- 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 of the Table.
- 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 of the Table.
- 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 of the Table.

Table 6. 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

Table 7. VALMA sum cloud+fallout+inhalation (Sv), median of one case values appearing at the distance, 99.5 % fractile of year 2012 cases.

	1 km	2 km	3 km	5 km	8 km	12 km
3.5 h	0.1840	0.0675	0.0431	0.0201	0.0099	0.0052
1 d	0.1850	0.0679	0.0433	0.0202	0.0099	0.0053
2 d	0.1860	0.0682	0.0435	0.0203	0.0100	0.0053
1 week	0.1880	0.0692	0.0441	0.0206	0.0101	0.0054
1 month	0.1940	0.0714	0.0456	0.0213	0.0105	0.0055
1 year	0.2100	0.0772	0.0492	0.0230	0.0114	0.0060

Table 8. VALMA sum cloud+fallout+inhalation (Sv), maximum 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	0.1200	0.0582	0.0365	0.0200	0.0109	0.0061
1 d	0.1200	0.0586	0.0367	0.0202	0.0110	0.0061
2 d	0.1210	0.0589	0.0369	0.0203	0.0111	0.0062
1 week	0.1220	0.0597	0.0374	0.0206	0.0112	0.0063
1 month	0.1270	0.0617	0.0387	0.0213	0.0116	0.0065
1 year	0.1370	0.0668	0.0418	0.0231	0.0125	0.0070

Table 9. VALMA sum cloud+fallout+inhalation (Sv), maximum of one case values appearing at the distance, 99.5 % fractile of year 2012 cases.

	1 km	2 km	3 km	5 km	8 km	12 km
3.5 h	0.3350	0.1480	0.0893	0.0421	0.0205	0.0109
1 d	0.3370	0.1480	0.0898	0.0423	0.0206	0.0109
2 d	0.3390	0.1490	0.0903	0.0426	0.0207	0.0110
1 week	0.3440	0.1510	0.0915	0.0431	0.0210	0.0111
1 month	0.3550	0.1560	0.0945	0.0446	0.0217	0.0115
1 year	0.3830	0.1690	0.1020	0.0481	0.0234	0.0124

9. IAEA / STUK criteria for protective countermeasures

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 off-site. In category II, such health effects are not expected off-site, and in category III, the need for protective actions is restricted to on-site only. The estimated potential off-site 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.

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

IAEA General Safety Guide GSG-2 (Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency) from 2011 (120 pages) is about 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

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

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

Various protective countermeasures usually have their criteria of onset defined as individual radiation dose to the members of the off-site 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.

The zones related to EPR (emergency preparedness & response) and longer-term actions can be explained as follows:

- 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 off-site 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.

In this work the emphasis is on the first two: PAZ and UPZ, which have biggest emergency significance.

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.

The recommendations in national guidelines, like the VAL guides of STUK (Finland) or NRC regulatory guides (PAG, protective action guidelines) generally are quite compliant with IAEA GSR Part 7. The STUK emergency guidelines (VAL) are the following:

- VAL-1: Protective measures in <u>early</u> phase of radiation emergency
  - Staying indoors, iodine pills, evacuation, etc.
- VAL-2: Protective measures in <u>late</u> 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

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 c_I > 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

In 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 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.

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

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 off-site doses. It was 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.

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, but some of them, 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.
- Communications to the general public: In many places the vast majority of people does not understand the units or levels of radiation-related quantities, and may have unnecessary fears of radiation.
- 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.

- Sizes of emergency planning zones and distances: 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.
- Preparedness also for very low probability events. These may include security and military events.
- Multiple units: At the same site, many units may be damaged by a common cause. In the case of SMR plants, there may be several small units at the site and possibly under the control of one common control room.
- 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.
- Involvement of the medical community.

10. International collaboration on NGR and EPR

It is suggested to 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 described in VTT-R-05548-16 [Hillberg et al. 2016].

10.1 International activities

In 2000, the IAEA established 'The international project on innovative nuclear reactors and fuel cycles' (INPRO). The purpose is to work towards sustainable energy production by nuclear power plants in the 21st century. Complementary to INPRO, the IAEA had or has currently several other programmes (Regular Budget Projects), the information on which can be found in www.iaea.org, but is reproduced here in compacted form for the reader's easy reference:

- Common issues and technologies for small and medium-sized reactors (2012-13)
- Near term & small and medium-sized reactor technology development (2014-17)
    - Technology roadmap for SMR deployments
    - Technical documents on
        - Design and operations of water-cooled SMRs
        - Human factor issues of multi-module SMR stations
        - Emergency planning zone (EPZ) and physical security requirements
    - E-toolkit for SMR technology assessment
    - Integral water-cooled reactor simulators
    - Non-electric engineered safety features
    - Molten salt cooled SMRs
    - Near-term water-cooled reactors and SMRs

Furthermore, the IAEA has established an SMR regulators' forum for better understanding of possible future challenges in SMR regulatory discussions.

Related to the EPZ technical documents mentioned above, there was be a meeting on emergency preparedness in Vienna in February 2017 ('IAEA Technical meeting on next generation reactors and emergency preparedness and response'). The most important developments in the meeting are described below based on VTT attendance in the meeting.

10.2 SMR emergency planning zones (EPZs)

In the design of SMRs, the inherent safety features are emphasized in most cases. The probability of melting of the fuel is calculated 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 off-site 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.

The main input needed for such SMR 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.

VTT (M Ilvonen) participated in the IAEA technical meeting (TM) of February 2017 on Gen4 emergency preparedness on behalf of the GENXFIN project (see 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 quite relevant for this work 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. EPR can be a design goal, but 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.

During the IAEA TM meeting, 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. 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 is participating in the new CRP I31029. In addition to VTT's own activities and contribution to the topic of the CRP, all participants' relevant activities will be reported within GENXFIN for the good of the SAFIR2018 research programme, and 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.

10.3 Definition and general information of the IAEA CRP I31029

New CRP (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

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.

According to Mr. Ramon de La Vega (Emergency Preparedness Coordinator of the IAEA Incident and Emergency Centre IEC), the new CRP I31029 is intended to start in January 2018 and be done by the end of 2020 (i.e. 3 years duration for the research). In the first Research Coordination Meeting to be organized soon after the start of the project (March 2018), the details for the Research Agreement to be signed by the IAEA and the participating research entities will be established. VTT will participate in the meeting.

11. Conclusions

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 off-site 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 terror 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 off-site emergency arrangements should be available. Their extent is a subject of intense debate between reactor developers and EPR people.

In this report the VTT-made VALMA code was used to assess the near-range off-site doses caused by release from an Olkiluoto-situated NuScale reactor unit. Real SILAM-calculated NWP-based weather data of year 2012 from the FMI was used. It must be noted however that 1 year of weather data is generally not enough to be representative; rather, e.g. 10 years should be available.

The inventory was acquired from an approximate Serpent calculation, but otherwise many conservative assumptions were used: high burnup for all assemblies, decontamination factor comparable to present large LWR (e.g. EPR), only 4 hours of cooling time between SCRAM and start of release, and all released into the atmosphere within 3 hours.

It would be too daring to try to use 'best estimates' with the resources budgeted for this work in GENXFIN, because there are simply too many unknowns in the NuScale design. Still, the conservative assumptions show that the off-site consequences are clearly lessened when compared with a large LWR.

It is very difficult to give any unambiguous recommendations on the EPZ size based on this work, but some considerations based on Tables 6-9 are given here (cf. Chapter 'IAEA / STUK criteria for protective countermeasures'):

- 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 of the Table.
- 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 of the Table.
- 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 of the Table.

For dose assessments, VALMA was used in this work. However, the straight-line Gaussian ARANO could rather be considered, because it calculates much faster than VALMA, and when the distances from source are in the near range (< 10 km), the simple weather model of ARANO should be well sufficient, particularly when the question is not about a single dispersion case but rather a vast number of them.

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.

The use of natural circulation and other passive safety features in NuScale, in both normal operation and safety systems, decreases the need of possibly malfunctioning active systems, but may in turn bring some unexpected complications in thermal-hydraulic behaviour of the plant. It is possible to analyse natural circulation with system codes (mainly 1D approach), but in cases with 3D effects proper analysis may have to include CFD simulations.

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 to Finland if the Finnish requirements are met. 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.

12. Summary

In this report the VTT-made VALMA code was used to assess the near-range off-site doses caused by release from an Olkiluoto-situated NuScale reactor unit. Real SILAM-calculated NWP-based weather data of year 2012 from the FMI was used. The inventory was acquired from an approximate Serpent calculation, but otherwise many conservative assumptions were used: high burnup for all assemblies, decontamination factor comparable to present large LWR (e.g. EPR), only 4 hours of cooling time between SCRAM and start of release, and all released into the atmosphere within 3 hours.

It would be too daring to try to use ‘best estimates’ with the resources budgeted for this work in GENXFIN, because there are simply too many unknowns in the NuScale design. Still, the conservative assumptions show that the off-site consequences are clearly lessened when compared with a large LWR.

In Finland, no new-build plant is acceptable without a feasible strategy for managing severe accidents. Differences from large power reactors include integrated RPVs, lower power levels and smaller reactor core radioactive inventories. Melt coolability and containment heat removal, among other topics, remain important issues.

A short introduction to severe accident management, the use of passive safety systems in SMRs and particularly the role of natural circulation in SMRs, in both normal operation and safety systems, was included. The use of natural circulation decreases the need of possibly malfunctioning active systems, but may in turn bring some unexpected complications in thermal-hydraulic behaviour of the plant. Proper analysis may have to include CFD simulations.

This project has enhanced knowledge on SMR designs and EPR (emergency preparedness and response) requirements, but it is not possible to give definite, unambiguous recommendations on the EPZ sizes based on this work only.

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

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Appendix: Results in tabulated form

In the table below, the most important information from all the ccdf curves (540 curves, when generated with all combinations of settings) is shown in compact form. The columns of the table are:

n	= running line number
time	= number of time step
	(70 = 3.5 h, 71 = 24 h, 72 = 48 h, 73 = 1 week, 74 = 1 month, 75 = 1 year)
dist	= distance (km) from release source
d	= dose pathway (1 cloud, 2 fallout, 3 inhalation, 4 direct sum, 5 shielded sum)
stat	= description of dose at the distance (mean, median or maximum value)
mean	= mean dose (Sv) of the year 2012 dispersion cases (8784 cases)
median	= median dose
95 %	= the 95 % fractile (i.e. only 5 % of doses exceed this value)
99.5 %	= the 99.5 % fractile
maximum	= maximum dose among the year 2012 dispersion cases

n	time	dist	d	stat	mean	median	95 %	99.5 %	maximum
1	70	1km	1	mean	4.26e-05	1.65e-05	1.91e-04	5.33e-04	9.56e-04
2	70	1km	1	med	4.38e-05	1.56e-05	1.97e-04	5.89e-04	1.00e-03
3	70	1km	1	max	8.41e-05	3.04e-05	3.77e-04	1.09e-03	2.40e-03
4	70	1km	2	mean	2.44e-05	1.01e-05	1.06e-04	2.90e-04	6.06e-04
5	70	1km	2	med	2.52e-05	9.30e-06	1.11e-04	3.38e-04	6.37e-04
6	70	1km	2	max	4.83e-05	1.87e-05	2.10e-04	5.75e-04	1.32e-03
7	70	1km	3	mean	1.35e-02	5.30e-03	5.99e-02	1.59e-01	2.75e-01
8	70	1km	3	med	1.39e-02	5.05e-03	6.19e-02	1.83e-01	3.10e-01
9	70	1km	3	max	2.66e-02	9.79e-03	1.19e-01	3.34e-01	6.43e-01
10	70	1km	4	mean	1.36e-02	5.33e-03	6.02e-02	1.60e-01	2.76e-01
11	70	1km	4	med	1.39e-02	5.08e-03	6.22e-02	1.84e-01	3.11e-01
12	70	1km	4	max	2.67e-02	9.83e-03	1.20e-01	3.36e-01	6.47e-01
13	70	1km	5	mean	1.35e-02	5.32e-03	6.01e-02	1.60e-01	2.76e-01
14	70	1km	5	med	1.39e-02	5.08e-03	6.22e-02	1.84e-01	3.11e-01
15	70	1km	5	max	2.67e-02	9.82e-03	1.20e-01	3.35e-01	6.46e-01
16	70	2km	1	mean	2.39e-05	1.20e-05	9.20e-05	2.02e-04	4.58e-04
17	70	2km	1	med	2.43e-05	1.25e-05	9.21e-05	2.17e-04	4.38e-04
18	70	2km	1	max	4.84e-05	2.31e-05	1.87e-04	4.92e-04	1.08e-03
19	70	2km	2	mean	1.37e-05	7.65e-06	4.84e-05	1.06e-04	2.21e-04
20	70	2km	2	med	1.41e-05	7.90e-06	5.15e-05	1.19e-04	2.43e-04
21	70	2km	2	max	2.78e-05	1.48e-05	1.01e-04	2.48e-04	4.85e-04
22	70	2km	3	mean	7.50e-03	3.84e-03	2.83e-02	6.07e-02	1.17e-01
23	70	2km	3	med	7.63e-03	4.00e-03	2.89e-02	6.72e-02	1.22e-01
24	70	2km	3	max	1.51e-02	7.43e-03	5.80e-02	1.47e-01	2.50e-01
25	70	2km	4	mean	7.54e-03	3.86e-03	2.85e-02	6.10e-02	1.17e-01
26	70	2km	4	med	7.67e-03	4.02e-03	2.90e-02	6.75e-02	1.23e-01
27	70	2km	4	max	1.52e-02	7.47e-03	5.83e-02	1.48e-01	2.51e-01
28	70	2km	5	mean	7.53e-03	3.86e-03	2.85e-02	6.09e-02	1.17e-01
29	70	2km	5	med	7.66e-03	4.02e-03	2.90e-02	6.75e-02	1.23e-01
30	70	2km	5	max	1.52e-02	7.46e-03	5.82e-02	1.48e-01	2.51e-01
31	70	3km	1	mean	1.48e-05	7.27e-06	5.71e-05	1.41e-04	2.71e-04
32	70	3km	1	med	1.48e-05	7.49e-06	5.68e-05	1.42e-04	2.46e-04
33	70	3km	1	max	3.02e-05	1.40e-05	1.19e-04	3.24e-04	6.40e-04
34	70	3km	2	mean	8.41e-06	4.77e-06	2.93e-05	7.05e-05	1.08e-04
35	70	3km	2	med	8.47e-06	4.83e-06	2.99e-05	7.21e-05	1.22e-04
36	70	3km	2	max	1.72e-05	9.18e-06	6.15e-05	1.55e-04	2.42e-04
37	70	3km	3	mean	4.58e-03	2.36e-03	1.72e-02	4.14e-02	6.31e-02

38	70	3km	3	med	4.61e-03	2.41e-03	1.71e-02	4.29e-02	6.16e-02
39	70	3km	3	max	9.31e-03	4.51e-03	3.63e-02	8.89e-02	1.41e-01
40	70	3km	4	mean	4.61e-03	2.37e-03	1.73e-02	4.16e-02	6.34e-02
41	70	3km	4	med	4.64e-03	2.42e-03	1.72e-02	4.31e-02	6.20e-02
42	70	3km	4	max	9.36e-03	4.53e-03	3.65e-02	8.94e-02	1.42e-01
43	70	3km	5	mean	4.60e-03	2.37e-03	1.73e-02	4.16e-02	6.34e-02
44	70	3km	5	med	4.63e-03	2.42e-03	1.72e-02	4.31e-02	6.19e-02
45	70	3km	5	max	9.35e-03	4.53e-03	3.65e-02	8.93e-02	1.42e-01
46	70	5km	1	mean	8.58e-06	4.32e-06	3.21e-05	7.60e-05	1.25e-04
47	70	5km	1	med	8.48e-06	4.44e-06	3.16e-05	7.40e-05	1.39e-04
48	70	5km	1	max	1.76e-05	8.40e-06	6.82e-05	1.62e-04	2.74e-04
49	70	5km	2	mean	4.77e-06	2.92e-06	1.56e-05	3.23e-05	4.90e-05
50	70	5km	2	med	4.74e-06	2.92e-06	1.57e-05	3.37e-05	6.33e-05
51	70	5km	2	max	9.83e-06	5.68e-06	3.35e-05	6.77e-05	9.53e-05
52	70	5km	3	mean	2.61e-03	1.40e-03	9.47e-03	1.98e-02	2.96e-02
53	70	5km	3	med	2.59e-03	1.43e-03	9.39e-03	2.00e-02	3.88e-02
54	70	5km	3	max	5.34e-03	2.72e-03	1.99e-02	4.19e-02	5.52e-02
55	70	5km	4	mean	2.62e-03	1.41e-03	9.52e-03	1.99e-02	2.98e-02
56	70	5km	4	med	2.61e-03	1.44e-03	9.44e-03	2.01e-02	3.90e-02
57	70	5km	4	max	5.36e-03	2.73e-03	2.00e-02	4.21e-02	5.56e-02
58	70	5km	5	mean	2.62e-03	1.41e-03	9.51e-03	1.99e-02	2.98e-02
59	70	5km	5	med	2.60e-03	1.44e-03	9.43e-03	2.01e-02	3.90e-02
60	70	5km	5	max	5.36e-03	2.73e-03	2.00e-02	4.21e-02	5.55e-02
61	70	8km	1	mean	5.05e-06	2.67e-06	1.87e-05	4.03e-05	6.45e-05
62	70	8km	1	med	5.06e-06	2.77e-06	1.87e-05	3.91e-05	7.70e-05
63	70	8km	1	max	1.03e-05	5.15e-06	3.88e-05	8.83e-05	1.33e-04
64	70	8km	2	mean	2.68e-06	1.74e-06	8.25e-06	1.54e-05	2.46e-05
65	70	8km	2	med	2.70e-06	1.76e-06	8.43e-06	1.63e-05	2.97e-05
66	70	8km	2	max	5.45e-06	3.44e-06	1.73e-05	3.26e-05	4.45e-05
67	70	8km	3	mean	1.50e-03	8.68e-04	5.23e-03	9.75e-03	1.60e-02
68	70	8km	3	med	1.51e-03	8.97e-04	5.19e-03	9.83e-03	1.92e-02
69	70	8km	3	max	3.02e-03	1.68e-03	1.09e-02	2.04e-02	2.79e-02
70	70	8km	4	mean	1.51e-03	8.73e-04	5.26e-03	9.80e-03	1.61e-02
71	70	8km	4	med	1.52e-03	9.02e-04	5.23e-03	9.88e-03	1.93e-02
72	70	8km	4	max	3.04e-03	1.69e-03	1.10e-02	2.05e-02	2.80e-02
73	70	8km	5	mean	1.50e-03	8.72e-04	5.25e-03	9.79e-03	1.61e-02
74	70	8km	5	med	1.52e-03	9.01e-04	5.22e-03	9.87e-03	1.92e-02
75	70	8km	5	max	3.04e-03	1.69e-03	1.09e-02	2.05e-02	2.80e-02
76	70	12km	1	mean	3.14e-06	1.77e-06	1.13e-05	2.34e-05	3.32e-05
77	70	12km	1	med	3.19e-06	1.85e-06	1.12e-05	2.34e-05	3.87e-05
78	70	12km	1	max	6.34e-06	3.42e-06	2.32e-05	5.02e-05	6.52e-05
79	70	12km	2	mean	1.55e-06	1.08e-06	4.45e-06	7.51e-06	1.08e-05
80	70	12km	2	med	1.58e-06	1.11e-06	4.50e-06	8.03e-06	1.24e-05
81	70	12km	2	max	3.13e-06	2.14e-06	9.23e-06	1.61e-05	2.13e-05
82	70	12km	3	mean	9.06e-04	5.81e-04	2.90e-03	4.95e-03	7.13e-03
83	70	12km	3	med	9.28e-04	6.05e-04	2.93e-03	5.20e-03	8.27e-03
84	70	12km	3	max	1.81e-03	1.12e-03	6.05e-03	1.08e-02	1.44e-02
85	70	12km	4	mean	9.10e-04	5.84e-04	2.92e-03	4.97e-03	7.17e-03
86	70	12km	4	med	9.33e-04	6.07e-04	2.94e-03	5.23e-03	8.32e-03
87	70	12km	4	max	1.82e-03	1.12e-03	6.09e-03	1.09e-02	1.45e-02
88	70	12km	5	mean	9.10e-04	5.83e-04	2.91e-03	4.97e-03	7.17e-03
89	70	12km	5	med	9.32e-04	6.07e-04	2.94e-03	5.23e-03	8.31e-03
90	70	12km	5	max	1.82e-03	1.12e-03	6.09e-03	1.09e-02	1.45e-02
91	71	1km	1	mean	4.24e-05	1.65e-05	1.91e-04	5.33e-04	9.56e-04
92	71	1km	1	med	4.35e-05	1.54e-05	1.96e-04	5.89e-04	1.00e-03
93	71	1km	1	max	8.41e-05	3.04e-05	3.77e-04	1.09e-03	2.40e-03
94	71	1km	2	mean	1.90e-04	7.88e-05	8.25e-04	2.18e-03	3.75e-03

95	71	1km	2	med	1.95e-04	7.38e-05	8.54e-04	2.51e-03	4.18e-03
96	71	1km	2	max	3.75e-04	1.46e-04	1.64e-03	4.54e-03	8.92e-03
97	71	1km	3	mean	1.34e-02	5.29e-03	5.94e-02	1.59e-01	2.75e-01
98	71	1km	3	med	1.38e-02	4.99e-03	6.13e-02	1.83e-01	3.10e-01
99	71	1km	3	max	2.66e-02	9.79e-03	1.19e-01	3.34e-01	6.43e-01
100	71	1km	4	mean	1.37e-02	5.38e-03	6.04e-02	1.62e-01	2.79e-01
101	71	1km	4	med	1.40e-02	5.08e-03	6.23e-02	1.86e-01	3.15e-01
102	71	1km	4	max	2.71e-02	9.96e-03	1.21e-01	3.40e-01	6.54e-01
103	71	1km	5	mean	1.36e-02	5.34e-03	6.00e-02	1.61e-01	2.78e-01
104	71	1km	5	med	1.39e-02	5.04e-03	6.19e-02	1.85e-01	3.13e-01
105	71	1km	5	max	2.69e-02	9.89e-03	1.20e-01	3.37e-01	6.50e-01
106	71	2km	1	mean	2.38e-05	1.19e-05	9.16e-05	2.02e-04	4.58e-04
107	71	2km	1	med	2.41e-05	1.24e-05	9.17e-05	2.17e-04	4.38e-04
108	71	2km	1	max	4.84e-05	2.31e-05	1.87e-04	4.92e-04	1.08e-03
109	71	2km	2	mean	1.09e-04	6.07e-05	3.90e-04	8.29e-04	1.58e-03
110	71	2km	2	med	1.11e-04	6.33e-05	4.05e-04	9.34e-04	1.67e-03
111	71	2km	2	max	2.20e-04	1.17e-04	8.04e-04	2.00e-03	3.39e-03
112	71	2km	3	mean	7.47e-03	3.83e-03	2.82e-02	6.07e-02	1.17e-01
113	71	2km	3	med	7.59e-03	3.97e-03	2.87e-02	6.72e-02	1.22e-01
114	71	2km	3	max	1.51e-02	7.43e-03	5.80e-02	1.47e-01	2.50e-01
115	71	2km	4	mean	7.60e-03	3.91e-03	2.87e-02	6.17e-02	1.19e-01
116	71	2km	4	med	7.72e-03	4.04e-03	2.92e-02	6.83e-02	1.24e-01
117	71	2km	4	max	1.54e-02	7.56e-03	5.90e-02	1.49e-01	2.54e-01
118	71	2km	5	mean	7.55e-03	3.87e-03	2.84e-02	6.13e-02	1.18e-01
119	71	2km	5	med	7.67e-03	4.01e-03	2.90e-02	6.79e-02	1.23e-01
120	71	2km	5	max	1.53e-02	7.51e-03	5.86e-02	1.48e-01	2.53e-01
121	71	3km	1	mean	1.47e-05	7.25e-06	5.67e-05	1.41e-04	2.71e-04
122	71	3km	1	med	1.47e-05	7.44e-06	5.56e-05	1.42e-04	2.46e-04
123	71	3km	1	max	3.02e-05	1.40e-05	1.19e-04	3.24e-04	6.40e-04
124	71	3km	2	mean	6.85e-05	3.87e-05	2.38e-04	5.65e-04	8.56e-04
125	71	3km	2	med	6.91e-05	3.95e-05	2.39e-04	5.74e-04	8.43e-04
126	71	3km	2	max	1.39e-04	7.46e-05	5.08e-04	1.21e-03	1.87e-03
127	71	3km	3	mean	4.56e-03	2.34e-03	1.71e-02	4.14e-02	6.31e-02
128	71	3km	3	med	4.58e-03	2.39e-03	1.69e-02	4.29e-02	6.16e-02
129	71	3km	3	max	9.31e-03	4.51e-03	3.63e-02	8.89e-02	1.41e-01
130	71	3km	4	mean	4.64e-03	2.39e-03	1.74e-02	4.21e-02	6.42e-02
131	71	3km	4	med	4.66e-03	2.44e-03	1.72e-02	4.36e-02	6.27e-02
132	71	3km	4	max	9.48e-03	4.60e-03	3.69e-02	9.04e-02	1.43e-01
133	71	3km	5	mean	4.61e-03	2.37e-03	1.73e-02	4.18e-02	6.38e-02
134	71	3km	5	med	4.63e-03	2.41e-03	1.71e-02	4.33e-02	6.23e-02
135	71	3km	5	max	9.41e-03	4.56e-03	3.67e-02	8.98e-02	1.42e-01
136	71	5km	1	mean	8.53e-06	4.30e-06	3.21e-05	7.60e-05	1.25e-04
137	71	5km	1	med	8.40e-06	4.39e-06	3.15e-05	7.40e-05	1.39e-04
138	71	5km	1	max	1.76e-05	8.40e-06	6.82e-05	1.62e-04	2.74e-04
139	71	5km	2	mean	4.06e-05	2.47e-05	1.33e-04	2.69e-04	4.01e-04
140	71	5km	2	med	4.05e-05	2.49e-05	1.33e-04	2.78e-04	5.24e-04
141	71	5km	2	max	8.29e-05	4.74e-05	2.81e-04	5.71e-04	7.40e-04
142	71	5km	3	mean	2.59e-03	1.40e-03	9.43e-03	1.98e-02	2.96e-02
143	71	5km	3	med	2.57e-03	1.42e-03	9.37e-03	2.00e-02	3.88e-02
144	71	5km	3	max	5.34e-03	2.72e-03	2.00e-02	4.19e-02	5.52e-02
145	71	5km	4	mean	2.64e-03	1.43e-03	9.60e-03	2.02e-02	3.01e-02
146	71	5km	4	med	2.62e-03	1.45e-03	9.53e-03	2.03e-02	3.95e-02
147	71	5km	4	max	5.44e-03	2.79e-03	2.03e-02	4.26e-02	5.62e-02
148	71	5km	5	mean	2.62e-03	1.42e-03	9.52e-03	2.00e-02	2.99e-02
149	71	5km	5	med	2.60e-03	1.43e-03	9.47e-03	2.02e-02	3.92e-02
150	71	5km	5	max	5.40e-03	2.76e-03	2.02e-02	4.23e-02	5.58e-02
151	71	8km	1	mean	5.02e-06	2.66e-06	1.86e-05	4.03e-05	6.45e-05

152	71	8km	1	med	5.00e-06	2.74e-06	1.85e-05	3.91e-05	7.70e-05
153	71	8km	1	max	1.03e-05	5.15e-06	3.88e-05	8.83e-05	1.33e-04
154	71	8km	2	mean	2.41e-05	1.59e-05	7.44e-05	1.33e-04	2.15e-04
155	71	8km	2	med	2.43e-05	1.61e-05	7.52e-05	1.39e-04	2.57e-04
156	71	8km	2	max	4.86e-05	3.11e-05	1.53e-04	2.78e-04	3.79e-04
157	71	8km	3	mean	1.49e-03	8.65e-04	5.18e-03	9.75e-03	1.60e-02
158	71	8km	3	med	1.50e-03	8.88e-04	5.16e-03	9.83e-03	1.92e-02
159	71	8km	3	max	3.03e-03	1.68e-03	1.09e-02	2.04e-02	2.79e-02
160	71	8km	4	mean	1.52e-03	8.86e-04	5.27e-03	9.92e-03	1.63e-02
161	71	8km	4	med	1.53e-03	9.09e-04	5.25e-03	9.99e-03	1.95e-02
162	71	8km	4	max	3.08e-03	1.72e-03	1.11e-02	2.07e-02	2.84e-02
163	71	8km	5	mean	1.51e-03	8.76e-04	5.23e-03	9.85e-03	1.62e-02
164	71	8km	5	med	1.51e-03	8.99e-04	5.21e-03	9.93e-03	1.94e-02
165	71	8km	5	max	3.06e-03	1.70e-03	1.10e-02	2.06e-02	2.82e-02
166	71	12km	1	mean	3.12e-06	1.77e-06	1.12e-05	2.34e-05	3.32e-05
167	71	12km	1	med	3.15e-06	1.84e-06	1.10e-05	2.34e-05	3.87e-05
168	71	12km	1	max	6.34e-06	3.42e-06	2.32e-05	5.02e-05	6.52e-05
169	71	12km	2	mean	1.48e-05	1.06e-05	4.13e-05	7.03e-05	9.56e-05
170	71	12km	2	med	1.51e-05	1.09e-05	4.22e-05	7.34e-05	1.10e-04
171	71	12km	2	max	2.97e-05	2.05e-05	8.71e-05	1.48e-04	1.93e-04
172	71	12km	3	mean	9.03e-04	5.82e-04	2.87e-03	4.95e-03	7.13e-03
173	71	12km	3	med	9.21e-04	6.01e-04	2.90e-03	5.20e-03	8.27e-03
174	71	12km	3	max	1.82e-03	1.12e-03	6.05e-03	1.08e-02	1.44e-02
175	71	12km	4	mean	9.20e-04	5.95e-04	2.92e-03	5.03e-03	7.26e-03
176	71	12km	4	med	9.39e-04	6.15e-04	2.95e-03	5.29e-03	8.42e-03
177	71	12km	4	max	1.85e-03	1.15e-03	6.18e-03	1.10e-02	1.46e-02
178	71	12km	5	mean	9.13e-04	5.90e-04	2.90e-03	5.00e-03	7.21e-03
179	71	12km	5	med	9.32e-04	6.08e-04	2.93e-03	5.26e-03	8.36e-03
180	71	12km	5	max	1.84e-03	1.14e-03	6.13e-03	1.09e-02	1.46e-02
181	72	1km	1	mean	4.24e-05	1.65e-05	1.91e-04	5.33e-04	9.56e-04
182	72	1km	1	med	4.35e-05	1.54e-05	1.96e-04	5.89e-04	1.00e-03
183	72	1km	1	max	8.41e-05	3.04e-05	3.77e-04	1.09e-03	2.40e-03
184	72	1km	2	mean	3.41e-04	1.41e-04	1.49e-03	3.91e-03	6.74e-03
185	72	1km	2	med	3.51e-04	1.33e-04	1.53e-03	4.52e-03	7.55e-03
186	72	1km	2	max	6.75e-04	2.63e-04	2.96e-03	8.19e-03	1.59e-02
187	72	1km	3	mean	1.34e-02	5.29e-03	5.94e-02	1.59e-01	2.75e-01
188	72	1km	3	med	1.38e-02	4.98e-03	6.13e-02	1.83e-01	3.10e-01
189	72	1km	3	max	2.66e-02	9.79e-03	1.19e-01	3.34e-01	6.43e-01
190	72	1km	4	mean	1.38e-02	5.44e-03	6.11e-02	1.64e-01	2.82e-01
191	72	1km	4	med	1.42e-02	5.14e-03	6.30e-02	1.88e-01	3.19e-01
192	72	1km	4	max	2.74e-02	1.01e-02	1.22e-01	3.43e-01	6.61e-01
193	72	1km	5	mean	1.36e-02	5.37e-03	6.04e-02	1.62e-01	2.79e-01
194	72	1km	5	med	1.40e-02	5.07e-03	6.22e-02	1.86e-01	3.15e-01
195	72	1km	5	max	2.70e-02	9.95e-03	1.21e-01	3.39e-01	6.53e-01
196	72	2km	1	mean	2.38e-05	1.19e-05	9.16e-05	2.02e-04	4.58e-04
197	72	2km	1	med	2.41e-05	1.24e-05	9.17e-05	2.17e-04	4.38e-04
198	72	2km	1	max	4.84e-05	2.31e-05	1.87e-04	4.92e-04	1.08e-03
199	72	2km	2	mean	1.96e-04	1.09e-04	7.03e-04	1.49e-03	2.85e-03
200	72	2km	2	med	2.00e-04	1.14e-04	7.26e-04	1.68e-03	3.00e-03
201	72	2km	2	max	3.96e-04	2.11e-04	1.45e-03	3.60e-03	6.11e-03
202	72	2km	3	mean	7.47e-03	3.83e-03	2.82e-02	6.07e-02	1.17e-01
203	72	2km	3	med	7.58e-03	3.96e-03	2.87e-02	6.72e-02	1.22e-01
204	72	2km	3	max	1.51e-02	7.43e-03	5.80e-02	1.47e-01	2.50e-01
205	72	2km	4	mean	7.69e-03	3.95e-03	2.90e-02	6.24e-02	1.20e-01
206	72	2km	4	med	7.81e-03	4.10e-03	2.96e-02	6.91e-02	1.25e-01
207	72	2km	4	max	1.56e-02	7.66e-03	5.96e-02	1.51e-01	2.57e-01
208	72	2km	5	mean	7.59e-03	3.90e-03	2.86e-02	6.16e-02	1.19e-01

209	72	2km	5	med	7.71e-03	4.03e-03	2.92e-02	6.82e-02	1.24e-01
210	72	2km	5	max	1.54e-02	7.55e-03	5.89e-02	1.49e-01	2.54e-01
211	72	3km	1	mean	1.47e-05	7.25e-06	5.67e-05	1.41e-04	2.71e-04
212	72	3km	1	med	1.47e-05	7.42e-06	5.54e-05	1.42e-04	2.46e-04
213	72	3km	1	max	3.02e-05	1.40e-05	1.19e-04	3.24e-04	6.40e-04
214	72	3km	2	mean	1.24e-04	6.97e-05	4.30e-04	1.02e-03	1.54e-03
215	72	3km	2	med	1.24e-04	7.12e-05	4.32e-04	1.05e-03	1.51e-03
216	72	3km	2	max	2.51e-04	1.34e-04	9.16e-04	2.18e-03	3.41e-03
217	72	3km	3	mean	4.56e-03	2.34e-03	1.71e-02	4.14e-02	6.31e-02
218	72	3km	3	med	4.57e-03	2.39e-03	1.69e-02	4.29e-02	6.16e-02
219	72	3km	3	max	9.31e-03	4.51e-03	3.63e-02	8.89e-02	1.41e-01
220	72	3km	4	mean	4.70e-03	2.42e-03	1.76e-02	4.25e-02	6.49e-02
221	72	3km	4	med	4.71e-03	2.47e-03	1.74e-02	4.40e-02	6.34e-02
222	72	3km	4	max	9.60e-03	4.66e-03	3.73e-02	9.14e-02	1.45e-01
223	72	3km	5	mean	4.63e-03	2.38e-03	1.74e-02	4.20e-02	6.41e-02
224	72	3km	5	med	4.65e-03	2.43e-03	1.71e-02	4.35e-02	6.26e-02
225	72	3km	5	max	9.47e-03	4.60e-03	3.69e-02	9.03e-02	1.43e-01
226	72	5km	1	mean	8.53e-06	4.29e-06	3.21e-05	7.60e-05	1.25e-04
227	72	5km	1	med	8.39e-06	4.38e-06	3.15e-05	7.40e-05	1.39e-04
228	72	5km	1	max	1.76e-05	8.40e-06	6.82e-05	1.62e-04	2.74e-04
229	72	5km	2	mean	7.33e-05	4.46e-05	2.40e-04	4.88e-04	7.23e-04
230	72	5km	2	med	7.31e-05	4.49e-05	2.40e-04	5.00e-04	9.47e-04
231	72	5km	2	max	1.50e-04	8.57e-05	5.07e-04	1.03e-03	1.34e-03
232	72	5km	3	mean	2.59e-03	1.40e-03	9.41e-03	1.98e-02	2.96e-02
233	72	5km	3	med	2.57e-03	1.42e-03	9.37e-03	2.00e-02	3.88e-02
234	72	5km	3	max	5.34e-03	2.72e-03	2.00e-02	4.19e-02	5.52e-02
235	72	5km	4	mean	2.67e-03	1.45e-03	9.69e-03	2.04e-02	3.05e-02
236	72	5km	4	med	2.65e-03	1.46e-03	9.64e-03	2.06e-02	3.99e-02
237	72	5km	4	max	5.51e-03	2.83e-03	2.05e-02	4.31e-02	5.68e-02
238	72	5km	5	mean	2.64e-03	1.43e-03	9.57e-03	2.01e-02	3.01e-02
239	72	5km	5	med	2.61e-03	1.44e-03	9.52e-03	2.03e-02	3.94e-02
240	72	5km	5	max	5.43e-03	2.78e-03	2.03e-02	4.26e-02	5.61e-02
241	72	8km	1	mean	5.02e-06	2.65e-06	1.86e-05	4.03e-05	6.45e-05
242	72	8km	1	med	4.99e-06	2.73e-06	1.85e-05	3.91e-05	7.70e-05
243	72	8km	1	max	1.03e-05	5.15e-06	3.88e-05	8.83e-05	1.33e-04
244	72	8km	2	mean	4.36e-05	2.88e-05	1.35e-04	2.41e-04	3.89e-04
245	72	8km	2	med	4.40e-05	2.91e-05	1.36e-04	2.53e-04	4.65e-04
246	72	8km	2	max	8.82e-05	5.64e-05	2.78e-04	5.05e-04	6.93e-04
247	72	8km	3	mean	1.49e-03	8.64e-04	5.18e-03	9.75e-03	1.60e-02
248	72	8km	3	med	1.49e-03	8.83e-04	5.16e-03	9.83e-03	1.92e-02
249	72	8km	3	max	3.03e-03	1.68e-03	1.09e-02	2.04e-02	2.79e-02
250	72	8km	4	mean	1.54e-03	8.96e-04	5.33e-03	1.00e-02	1.65e-02
251	72	8km	4	med	1.54e-03	9.20e-04	5.31e-03	1.01e-02	1.97e-02
252	72	8km	4	max	3.12e-03	1.74e-03	1.12e-02	2.10e-02	2.87e-02
253	72	8km	5	mean	1.52e-03	8.81e-04	5.26e-03	9.90e-03	1.63e-02
254	72	8km	5	med	1.52e-03	9.02e-04	5.24e-03	9.98e-03	1.95e-02
255	72	8km	5	max	3.08e-03	1.72e-03	1.11e-02	2.07e-02	2.84e-02
256	72	12km	1	mean	3.12e-06	1.77e-06	1.12e-05	2.34e-05	3.32e-05
257	72	12km	1	med	3.14e-06	1.83e-06	1.10e-05	2.34e-05	3.87e-05
258	72	12km	1	max	6.34e-06	3.42e-06	2.32e-05	5.02e-05	6.52e-05
259	72	12km	2	mean	2.69e-05	1.92e-05	7.50e-05	1.27e-04	1.73e-04
260	72	12km	2	med	2.75e-05	1.99e-05	7.69e-05	1.33e-04	2.00e-04
261	72	12km	2	max	5.41e-05	3.75e-05	1.58e-04	2.70e-04	3.50e-04
262	72	12km	3	mean	9.01e-04	5.81e-04	2.87e-03	4.95e-03	7.13e-03
263	72	12km	3	med	9.19e-04	6.00e-04	2.90e-03	5.20e-03	8.27e-03
264	72	12km	3	max	1.82e-03	1.12e-03	6.05e-03	1.08e-02	1.44e-02
265	72	12km	4	mean	9.31e-04	6.02e-04	2.95e-03	5.09e-03	7.33e-03

266	72	12km	4	med	9.49e-04	6.23e-04	2.98e-03	5.35e-03	8.51e-03
267	72	12km	4	max	1.88e-03	1.16e-03	6.25e-03	1.11e-02	1.48e-02
268	72	12km	5	mean	9.18e-04	5.92e-04	2.92e-03	5.03e-03	7.25e-03
269	72	12km	5	med	9.35e-04	6.11e-04	2.94e-03	5.29e-03	8.41e-03
270	72	12km	5	max	1.85e-03	1.15e-03	6.17e-03	1.10e-02	1.46e-02
271	73	1km	1	mean	4.24e-05	1.65e-05	1.91e-04	5.33e-04	9.56e-04
272	73	1km	1	med	4.35e-05	1.54e-05	1.96e-04	5.89e-04	1.00e-03
273	73	1km	1	max	8.41e-05	3.04e-05	3.77e-04	1.09e-03	2.40e-03
274	73	1km	2	mean	7.19e-04	2.97e-04	3.14e-03	8.25e-03	1.42e-02
275	73	1km	2	med	7.38e-04	2.80e-04	3.22e-03	9.45e-03	1.59e-02
276	73	1km	2	max	1.42e-03	5.54e-04	6.21e-03	1.72e-02	3.34e-02
277	73	1km	3	mean	1.34e-02	5.29e-03	5.94e-02	1.59e-01	2.75e-01
278	73	1km	3	med	1.38e-02	4.98e-03	6.13e-02	1.83e-01	3.10e-01
279	73	1km	3	max	2.66e-02	9.79e-03	1.19e-01	3.34e-01	6.43e-01
280	73	1km	4	mean	1.42e-02	5.60e-03	6.27e-02	1.68e-01	2.90e-01
281	73	1km	4	med	1.46e-02	5.29e-03	6.47e-02	1.93e-01	3.27e-01
282	73	1km	4	max	2.81e-02	1.04e-02	1.26e-01	3.52e-01	6.79e-01
283	73	1km	5	mean	1.38e-02	5.44e-03	6.12e-02	1.64e-01	2.83e-01
284	73	1km	5	med	1.42e-02	5.15e-03	6.31e-02	1.88e-01	3.19e-01
285	73	1km	5	max	2.74e-02	1.01e-02	1.22e-01	3.44e-01	6.62e-01
286	73	2km	1	mean	2.38e-05	1.19e-05	9.16e-05	2.02e-04	4.58e-04
287	73	2km	1	med	2.41e-05	1.24e-05	9.17e-05	2.17e-04	4.38e-04
288	73	2km	1	max	4.84e-05	2.31e-05	1.87e-04	4.92e-04	1.08e-03
289	73	2km	2	mean	4.14e-04	2.30e-04	1.48e-03	3.15e-03	6.01e-03
290	73	2km	2	med	4.21e-04	2.40e-04	1.53e-03	3.52e-03	6.31e-03
291	73	2km	2	max	8.34e-04	4.43e-04	3.07e-03	7.60e-03	1.29e-02
292	73	2km	3	mean	7.47e-03	3.83e-03	2.82e-02	6.07e-02	1.17e-01
293	73	2km	3	med	7.58e-03	3.96e-03	2.87e-02	6.72e-02	1.22e-01
294	73	2km	3	max	1.51e-02	7.43e-03	5.80e-02	1.47e-01	2.50e-01
295	73	2km	4	mean	7.91e-03	4.08e-03	2.98e-02	6.40e-02	1.23e-01
296	73	2km	4	med	8.03e-03	4.23e-03	3.04e-02	7.09e-02	1.29e-01
297	73	2km	4	max	1.60e-02	7.90e-03	6.12e-02	1.55e-01	2.64e-01
298	73	2km	5	mean	7.70e-03	3.96e-03	2.90e-02	6.24e-02	1.20e-01
299	73	2km	5	med	7.82e-03	4.10e-03	2.96e-02	6.92e-02	1.26e-01
300	73	2km	5	max	1.56e-02	7.68e-03	5.97e-02	1.51e-01	2.57e-01
301	73	3km	1	mean	1.47e-05	7.25e-06	5.67e-05	1.41e-04	2.71e-04
302	73	3km	1	med	1.47e-05	7.42e-06	5.54e-05	1.42e-04	2.46e-04
303	73	3km	1	max	3.02e-05	1.40e-05	1.19e-04	3.24e-04	6.40e-04
304	73	3km	2	mean	2.60e-04	1.47e-04	9.04e-04	2.15e-03	3.25e-03
305	73	3km	2	med	2.62e-04	1.50e-04	9.10e-04	2.22e-03	3.18e-03
306	73	3km	2	max	5.29e-04	2.83e-04	1.93e-03	4.60e-03	7.22e-03
307	73	3km	3	mean	4.56e-03	2.34e-03	1.71e-02	4.14e-02	6.31e-02
308	73	3km	3	med	4.57e-03	2.39e-03	1.69e-02	4.29e-02	6.16e-02
309	73	3km	3	max	9.31e-03	4.51e-03	3.63e-02	8.89e-02	1.41e-01
310	73	3km	4	mean	4.83e-03	2.50e-03	1.81e-02	4.37e-02	6.66e-02
311	73	3km	4	med	4.85e-03	2.55e-03	1.78e-02	4.52e-02	6.50e-02
312	73	3km	4	max	9.87e-03	4.82e-03	3.84e-02	9.38e-02	1.49e-01
313	73	3km	5	mean	4.70e-03	2.43e-03	1.76e-02	4.26e-02	6.50e-02
314	73	3km	5	med	4.72e-03	2.47e-03	1.74e-02	4.41e-02	6.34e-02
315	73	3km	5	max	9.61e-03	4.67e-03	3.74e-02	9.15e-02	1.45e-01
316	73	5km	1	mean	8.53e-06	4.29e-06	3.21e-05	7.60e-05	1.25e-04
317	73	5km	1	med	8.39e-06	4.38e-06	3.15e-05	7.40e-05	1.39e-04
318	73	5km	1	max	1.76e-05	8.40e-06	6.82e-05	1.62e-04	2.74e-04
319	73	5km	2	mean	1.55e-04	9.40e-05	5.07e-04	1.03e-03	1.53e-03
320	73	5km	2	med	1.54e-04	9.48e-05	5.08e-04	1.06e-03	2.00e-03
321	73	5km	2	max	3.17e-04	1.81e-04	1.07e-03	2.16e-03	2.84e-03
322	73	5km	3	mean	2.59e-03	1.40e-03	9.41e-03	1.98e-02	2.96e-02

323	73	5km	3	med	2.57e-03	1.42e-03	9.37e-03	2.00e-02	3.88e-02
324	73	5km	3	max	5.34e-03	2.72e-03	2.00e-02	4.19e-02	5.52e-02
325	73	5km	4	mean	2.76e-03	1.50e-03	9.95e-03	2.10e-02	3.13e-02
326	73	5km	4	med	2.73e-03	1.51e-03	9.92e-03	2.11e-02	4.09e-02
327	73	5km	4	max	5.67e-03	2.93e-03	2.11e-02	4.42e-02	5.83e-02
328	73	5km	5	mean	2.68e-03	1.45e-03	9.70e-03	2.04e-02	3.05e-02
329	73	5km	5	med	2.65e-03	1.47e-03	9.65e-03	2.06e-02	3.99e-02
330	73	5km	5	max	5.51e-03	2.83e-03	2.06e-02	4.31e-02	5.69e-02
331	73	8km	1	mean	5.02e-06	2.65e-06	1.86e-05	4.03e-05	6.45e-05
332	73	8km	1	med	4.99e-06	2.73e-06	1.85e-05	3.91e-05	7.70e-05
333	73	8km	1	max	1.03e-05	5.15e-06	3.88e-05	8.83e-05	1.33e-04
334	73	8km	2	mean	9.23e-05	6.10e-05	2.86e-04	5.10e-04	8.24e-04
335	73	8km	2	med	9.31e-05	6.17e-05	2.88e-04	5.35e-04	9.81e-04
336	73	8km	2	max	1.87e-04	1.19e-04	5.91e-04	1.07e-03	1.47e-03
337	73	8km	3	mean	1.49e-03	8.64e-04	5.18e-03	9.75e-03	1.60e-02
338	73	8km	3	med	1.49e-03	8.83e-04	5.16e-03	9.83e-03	1.92e-02
339	73	8km	3	max	3.03e-03	1.68e-03	1.09e-02	2.04e-02	2.79e-02
340	73	8km	4	mean	1.59e-03	9.30e-04	5.47e-03	1.03e-02	1.69e-02
341	73	8km	4	med	1.59e-03	9.57e-04	5.47e-03	1.04e-02	2.02e-02
342	73	8km	4	max	3.22e-03	1.81e-03	1.15e-02	2.15e-02	2.95e-02
343	73	8km	5	mean	1.54e-03	8.98e-04	5.33e-03	1.00e-02	1.65e-02
344	73	8km	5	med	1.55e-03	9.22e-04	5.32e-03	1.01e-02	1.97e-02
345	73	8km	5	max	3.13e-03	1.75e-03	1.12e-02	2.10e-02	2.87e-02
346	73	12km	1	mean	3.12e-06	1.77e-06	1.12e-05	2.34e-05	3.32e-05
347	73	12km	1	med	3.14e-06	1.83e-06	1.10e-05	2.34e-05	3.87e-05
348	73	12km	1	max	6.34e-06	3.42e-06	2.32e-05	5.02e-05	6.52e-05
349	73	12km	2	mean	5.71e-05	4.08e-05	1.59e-04	2.71e-04	3.66e-04
350	73	12km	2	med	5.82e-05	4.21e-05	1.63e-04	2.83e-04	4.24e-04
351	73	12km	2	max	1.15e-04	7.95e-05	3.36e-04	5.73e-04	7.42e-04
352	73	12km	3	mean	9.01e-04	5.81e-04	2.87e-03	4.95e-03	7.13e-03
353	73	12km	3	med	9.19e-04	6.00e-04	2.90e-03	5.20e-03	8.27e-03
354	73	12km	3	max	1.82e-03	1.12e-03	6.05e-03	1.08e-02	1.44e-02
355	73	12km	4	mean	9.62e-04	6.22e-04	3.03e-03	5.22e-03	7.53e-03
356	73	12km	4	med	9.80e-04	6.47e-04	3.06e-03	5.50e-03	8.73e-03
357	73	12km	4	max	1.94e-03	1.21e-03	6.42e-03	1.14e-02	1.52e-02
358	73	12km	5	mean	9.33e-04	6.03e-04	2.96e-03	5.09e-03	7.34e-03
359	73	12km	5	med	9.51e-04	6.24e-04	2.98e-03	5.37e-03	8.52e-03
360	73	12km	5	max	1.88e-03	1.16e-03	6.26e-03	1.11e-02	1.48e-02
361	74	1km	1	mean	4.24e-05	1.65e-05	1.91e-04	5.33e-04	9.56e-04
362	74	1km	1	med	4.35e-05	1.54e-05	1.96e-04	5.89e-04	1.00e-03
363	74	1km	1	max	8.41e-05	3.04e-05	3.77e-04	1.09e-03	2.40e-03
364	74	1km	2	mean	1.67e-03	6.89e-04	7.28e-03	1.91e-02	3.29e-02
365	74	1km	2	med	1.71e-03	6.49e-04	7.46e-03	2.19e-02	3.70e-02
366	74	1km	2	max	3.30e-03	1.28e-03	1.44e-02	3.99e-02	7.71e-02
367	74	1km	3	mean	1.34e-02	5.29e-03	5.94e-02	1.59e-01	2.75e-01
368	74	1km	3	med	1.38e-02	4.98e-03	6.13e-02	1.83e-01	3.10e-01
369	74	1km	3	max	2.66e-02	9.79e-03	1.19e-01	3.34e-01	6.43e-01
370	74	1km	4	mean	1.51e-02	6.01e-03	6.69e-02	1.79e-01	3.09e-01
371	74	1km	4	med	1.55e-02	5.68e-03	6.88e-02	2.05e-01	3.48e-01
372	74	1km	4	max	3.00e-02	1.11e-02	1.34e-01	3.75e-01	7.23e-01
373	74	1km	5	mean	1.43e-02	5.65e-03	6.32e-02	1.69e-01	2.92e-01
374	74	1km	5	med	1.47e-02	5.34e-03	6.52e-02	1.94e-01	3.29e-01
375	74	1km	5	max	2.83e-02	1.04e-02	1.27e-01	3.55e-01	6.84e-01
376	74	2km	1	mean	2.38e-05	1.19e-05	9.16e-05	2.02e-04	4.58e-04
377	74	2km	1	med	2.41e-05	1.24e-05	9.17e-05	2.17e-04	4.38e-04
378	74	2km	1	max	4.84e-05	2.31e-05	1.87e-04	4.92e-04	1.08e-03
379	74	2km	2	mean	9.60e-04	5.34e-04	3.43e-03	7.31e-03	1.40e-02

380	74	2km	2	med	9.78e-04	5.57e-04	3.54e-03	8.16e-03	1.46e-02
381	74	2km	2	max	1.94e-03	1.03e-03	7.12e-03	1.76e-02	2.99e-02
382	74	2km	3	mean	7.47e-03	3.83e-03	2.82e-02	6.07e-02	1.17e-01
383	74	2km	3	med	7.58e-03	3.96e-03	2.87e-02	6.72e-02	1.22e-01
384	74	2km	3	max	1.51e-02	7.43e-03	5.80e-02	1.47e-01	2.50e-01
385	74	2km	4	mean	8.45e-03	4.38e-03	3.17e-02	6.81e-02	1.31e-01
386	74	2km	4	med	8.59e-03	4.55e-03	3.24e-02	7.55e-02	1.37e-01
387	74	2km	4	max	1.71e-02	8.50e-03	6.52e-02	1.65e-01	2.81e-01
388	74	2km	5	mean	7.97e-03	4.12e-03	3.00e-02	6.45e-02	1.24e-01
389	74	2km	5	med	8.10e-03	4.27e-03	3.06e-02	7.14e-02	1.30e-01
390	74	2km	5	max	1.62e-02	7.97e-03	6.17e-02	1.56e-01	2.66e-01
391	74	3km	1	mean	1.47e-05	7.25e-06	5.67e-05	1.41e-04	2.71e-04
392	74	3km	1	med	1.47e-05	7.42e-06	5.54e-05	1.42e-04	2.46e-04
393	74	3km	1	max	3.02e-05	1.40e-05	1.19e-04	3.24e-04	6.40e-04
394	74	3km	2	mean	6.04e-04	3.41e-04	2.10e-03	4.98e-03	7.54e-03
395	74	3km	2	med	6.09e-04	3.47e-04	2.11e-03	5.18e-03	7.38e-03
396	74	3km	2	max	1.23e-03	6.57e-04	4.49e-03	1.06e-02	1.68e-02
397	74	3km	3	mean	4.56e-03	2.34e-03	1.71e-02	4.14e-02	6.31e-02
398	74	3km	3	med	4.57e-03	2.39e-03	1.69e-02	4.29e-02	6.16e-02
399	74	3km	3	max	9.31e-03	4.51e-03	3.63e-02	8.89e-02	1.41e-01
400	74	3km	4	mean	5.18e-03	2.71e-03	1.93e-02	4.65e-02	7.09e-02
401	74	3km	4	med	5.20e-03	2.75e-03	1.90e-02	4.81e-02	6.92e-02
402	74	3km	4	max	1.06e-02	5.21e-03	4.08e-02	9.98e-02	1.58e-01
403	74	3km	5	mean	4.87e-03	2.53e-03	1.82e-02	4.40e-02	6.71e-02
404	74	3km	5	med	4.89e-03	2.57e-03	1.80e-02	4.56e-02	6.55e-02
405	74	3km	5	max	9.96e-03	4.87e-03	3.87e-02	9.45e-02	1.50e-01
406	74	5km	1	mean	8.53e-06	4.29e-06	3.21e-05	7.60e-05	1.25e-04
407	74	5km	1	med	8.39e-06	4.38e-06	3.15e-05	7.40e-05	1.39e-04
408	74	5km	1	max	1.76e-05	8.40e-06	6.82e-05	1.62e-04	2.74e-04
409	74	5km	2	mean	3.60e-04	2.18e-04	1.18e-03	2.41e-03	3.54e-03
410	74	5km	2	med	3.59e-04	2.20e-04	1.18e-03	2.47e-03	4.64e-03
411	74	5km	2	max	7.36e-04	4.20e-04	2.50e-03	5.01e-03	6.59e-03
412	74	5km	3	mean	2.59e-03	1.40e-03	9.41e-03	1.98e-02	2.96e-02
413	74	5km	3	med	2.57e-03	1.42e-03	9.37e-03	2.00e-02	3.88e-02
414	74	5km	3	max	5.34e-03	2.72e-03	2.00e-02	4.19e-02	5.52e-02
415	74	5km	4	mean	2.96e-03	1.63e-03	1.06e-02	2.24e-02	3.33e-02
416	74	5km	4	med	2.94e-03	1.65e-03	1.06e-02	2.25e-02	4.36e-02
417	74	5km	4	max	6.09e-03	3.21e-03	2.25e-02	4.71e-02	6.21e-02
418	74	5km	5	mean	2.78e-03	1.52e-03	1.00e-02	2.12e-02	3.15e-02
419	74	5km	5	med	2.76e-03	1.53e-03	1.00e-02	2.13e-02	4.13e-02
420	74	5km	5	max	5.72e-03	2.96e-03	2.13e-02	4.46e-02	5.88e-02
421	74	8km	1	mean	5.02e-06	2.65e-06	1.86e-05	4.03e-05	6.45e-05
422	74	8km	1	med	4.99e-06	2.73e-06	1.85e-05	3.91e-05	7.70e-05
423	74	8km	1	max	1.03e-05	5.15e-06	3.88e-05	8.83e-05	1.33e-04
424	74	8km	2	mean	2.15e-04	1.42e-04	6.64e-04	1.19e-03	1.91e-03
425	74	8km	2	med	2.16e-04	1.44e-04	6.73e-04	1.24e-03	2.28e-03
426	74	8km	2	max	4.34e-04	2.77e-04	1.37e-03	2.49e-03	3.43e-03
427	74	8km	3	mean	1.49e-03	8.64e-04	5.18e-03	9.75e-03	1.60e-02
428	74	8km	3	med	1.49e-03	8.83e-04	5.16e-03	9.83e-03	1.92e-02
429	74	8km	3	max	3.03e-03	1.68e-03	1.09e-02	2.04e-02	2.79e-02
430	74	8km	4	mean	1.71e-03	1.02e-03	5.85e-03	1.10e-02	1.80e-02
431	74	8km	4	med	1.72e-03	1.05e-03	5.86e-03	1.12e-02	2.15e-02
432	74	8km	4	max	3.47e-03	1.97e-03	1.23e-02	2.29e-02	3.14e-02
433	74	8km	5	mean	1.60e-03	9.42e-04	5.52e-03	1.04e-02	1.70e-02
434	74	8km	5	med	1.61e-03	9.67e-04	5.52e-03	1.05e-02	2.04e-02
435	74	8km	5	max	3.25e-03	1.83e-03	1.16e-02	2.17e-02	2.97e-02
436	74	12km	1	mean	3.12e-06	1.77e-06	1.12e-05	2.34e-05	3.32e-05

437	74	12km	1	med	3.14e-06	1.83e-06	1.10e-05	2.34e-05	3.87e-05
438	74	12km	1	max	6.34e-06	3.42e-06	2.32e-05	5.02e-05	6.52e-05
439	74	12km	2	mean	1.33e-04	9.51e-05	3.70e-04	6.30e-04	8.53e-04
440	74	12km	2	med	1.36e-04	9.80e-05	3.79e-04	6.59e-04	9.88e-04
441	74	12km	2	max	2.67e-04	1.85e-04	7.83e-04	1.34e-03	1.73e-03
442	74	12km	3	mean	9.01e-04	5.81e-04	2.87e-03	4.95e-03	7.13e-03
443	74	12km	3	med	9.19e-04	6.00e-04	2.90e-03	5.20e-03	8.27e-03
444	74	12km	3	max	1.82e-03	1.12e-03	6.05e-03	1.08e-02	1.44e-02
445	74	12km	4	mean	1.04e-03	6.79e-04	3.24e-03	5.56e-03	8.01e-03
446	74	12km	4	med	1.06e-03	7.09e-04	3.27e-03	5.86e-03	9.30e-03
447	74	12km	4	max	2.09e-03	1.32e-03	6.84e-03	1.21e-02	1.62e-02
448	74	12km	5	mean	9.71e-04	6.29e-04	3.06e-03	5.26e-03	7.59e-03
449	74	12km	5	med	9.90e-04	6.55e-04	3.09e-03	5.54e-03	8.80e-03
450	74	12km	5	max	1.96e-03	1.22e-03	6.48e-03	1.15e-02	1.53e-02
451	75	1km	1	mean	4.24e-05	1.65e-05	1.91e-04	5.33e-04	9.56e-04
452	75	1km	1	med	4.35e-05	1.54e-05	1.96e-04	5.89e-04	1.00e-03
453	75	1km	1	max	8.41e-05	3.04e-05	3.77e-04	1.09e-03	2.40e-03
454	75	1km	2	mean	4.03e-03	1.67e-03	1.76e-02	4.62e-02	7.97e-02
455	75	1km	2	med	4.14e-03	1.57e-03	1.81e-02	5.29e-02	8.97e-02
456	75	1km	2	max	7.98e-03	3.11e-03	3.48e-02	9.67e-02	1.87e-01
457	75	1km	3	mean	1.34e-02	5.29e-03	5.94e-02	1.59e-01	2.75e-01
458	75	1km	3	med	1.38e-02	4.98e-03	6.13e-02	1.83e-01	3.10e-01
459	75	1km	3	max	2.66e-02	9.79e-03	1.19e-01	3.34e-01	6.43e-01
460	75	1km	4	mean	1.75e-02	7.01e-03	7.75e-02	2.06e-01	3.55e-01
461	75	1km	4	med	1.80e-02	6.62e-03	7.96e-02	2.36e-01	4.01e-01
462	75	1km	4	max	3.47e-02	1.29e-02	1.54e-01	4.32e-01	8.32e-01
463	75	1km	5	mean	1.55e-02	6.15e-03	6.84e-02	1.83e-01	3.16e-01
464	75	1km	5	med	1.59e-02	5.82e-03	7.04e-02	2.10e-01	3.56e-01
465	75	1km	5	max	3.07e-02	1.14e-02	1.37e-01	3.83e-01	7.39e-01
466	75	2km	1	mean	2.38e-05	1.19e-05	9.16e-05	2.02e-04	4.58e-04
467	75	2km	1	med	2.41e-05	1.24e-05	9.17e-05	2.17e-04	4.38e-04
468	75	2km	1	max	4.84e-05	2.31e-05	1.87e-04	4.92e-04	1.08e-03
469	75	2km	2	mean	2.32e-03	1.29e-03	8.32e-03	1.77e-02	3.38e-02
470	75	2km	2	med	2.37e-03	1.35e-03	8.58e-03	1.98e-02	3.54e-02
471	75	2km	2	max	4.69e-03	2.49e-03	1.73e-02	4.27e-02	7.23e-02
472	75	2km	3	mean	7.47e-03	3.83e-03	2.82e-02	6.07e-02	1.17e-01
473	75	2km	3	med	7.58e-03	3.96e-03	2.87e-02	6.72e-02	1.22e-01
474	75	2km	3	max	1.51e-02	7.43e-03	5.80e-02	1.47e-01	2.50e-01
475	75	2km	4	mean	9.82e-03	5.14e-03	3.67e-02	7.85e-02	1.51e-01
476	75	2km	4	med	9.98e-03	5.36e-03	3.74e-02	8.75e-02	1.58e-01
477	75	2km	4	max	1.99e-02	9.95e-03	7.54e-02	1.90e-01	3.23e-01
478	75	2km	5	mean	8.65e-03	4.50e-03	3.25e-02	6.97e-02	1.34e-01
479	75	2km	5	med	8.79e-03	4.66e-03	3.31e-02	7.72e-02	1.40e-01
480	75	2km	5	max	1.75e-02	8.72e-03	6.68e-02	1.69e-01	2.87e-01
481	75	3km	1	mean	1.47e-05	7.25e-06	5.67e-05	1.41e-04	2.71e-04
482	75	3km	1	med	1.47e-05	7.42e-06	5.54e-05	1.42e-04	2.46e-04
483	75	3km	1	max	3.02e-05	1.40e-05	1.19e-04	3.24e-04	6.40e-04
484	75	3km	2	mean	1.46e-03	8.25e-04	5.08e-03	1.21e-02	1.83e-02
485	75	3km	2	med	1.47e-03	8.41e-04	5.11e-03	1.26e-02	1.79e-02
486	75	3km	2	max	2.98e-03	1.59e-03	1.09e-02	2.58e-02	4.08e-02
487	75	3km	3	mean	4.56e-03	2.34e-03	1.71e-02	4.14e-02	6.31e-02
488	75	3km	3	med	4.57e-03	2.39e-03	1.69e-02	4.29e-02	6.16e-02
489	75	3km	3	max	9.31e-03	4.51e-03	3.63e-02	8.89e-02	1.41e-01
490	75	3km	4	mean	6.04e-03	3.18e-03	2.22e-02	5.35e-02	8.16e-02
491	75	3km	4	med	6.06e-03	3.25e-03	2.19e-02	5.54e-02	7.98e-02
492	75	3km	4	max	1.23e-02	6.15e-03	4.70e-02	1.15e-01	1.82e-01
493	75	3km	5	mean	5.30e-03	2.78e-03	1.97e-02	4.75e-02	7.25e-02

494	75	3km	5	med	5.32e-03	2.83e-03	1.94e-02	4.92e-02	7.08e-02
495	75	3km	5	max	1.08e-02	5.34e-03	4.18e-02	1.02e-01	1.62e-01
496	75	5km	1	mean	8.53e-06	4.29e-06	3.21e-05	7.60e-05	1.25e-04
497	75	5km	1	med	8.39e-06	4.38e-06	3.15e-05	7.40e-05	1.39e-04
498	75	5km	1	max	1.76e-05	8.40e-06	6.82e-05	1.62e-04	2.74e-04
499	75	5km	2	mean	8.72e-04	5.29e-04	2.86e-03	5.83e-03	8.59e-03
500	75	5km	2	med	8.69e-04	5.33e-04	2.86e-03	5.98e-03	1.13e-02
501	75	5km	2	max	1.78e-03	1.02e-03	6.05e-03	1.21e-02	1.60e-02
502	75	5km	3	mean	2.59e-03	1.40e-03	9.41e-03	1.98e-02	2.96e-02
503	75	5km	3	med	2.57e-03	1.42e-03	9.37e-03	2.00e-02	3.88e-02
504	75	5km	3	max	5.34e-03	2.72e-03	2.00e-02	4.19e-02	5.52e-02
505	75	5km	4	mean	3.47e-03	1.96e-03	1.23e-02	2.57e-02	3.83e-02
506	75	5km	4	med	3.45e-03	1.99e-03	1.22e-02	2.61e-02	5.02e-02
507	75	5km	4	max	7.14e-03	3.84e-03	2.60e-02	5.42e-02	7.15e-02
508	75	5km	5	mean	3.04e-03	1.68e-03	1.08e-02	2.29e-02	3.40e-02
509	75	5km	5	med	3.01e-03	1.70e-03	1.08e-02	2.30e-02	4.46e-02
510	75	5km	5	max	6.25e-03	3.31e-03	2.31e-02	4.81e-02	6.35e-02
511	75	8km	1	mean	5.02e-06	2.65e-06	1.86e-05	4.03e-05	6.45e-05
512	75	8km	1	med	4.99e-06	2.73e-06	1.85e-05	3.91e-05	7.70e-05
513	75	8km	1	max	1.03e-05	5.15e-06	3.88e-05	8.83e-05	1.33e-04
514	75	8km	2	mean	5.21e-04	3.44e-04	1.61e-03	2.89e-03	4.64e-03
515	75	8km	2	med	5.25e-04	3.48e-04	1.63e-03	3.01e-03	5.54e-03
516	75	8km	2	max	1.05e-03	6.72e-04	3.33e-03	6.03e-03	8.33e-03
517	75	8km	3	mean	1.49e-03	8.64e-04	5.18e-03	9.75e-03	1.60e-02
518	75	8km	3	med	1.49e-03	8.83e-04	5.16e-03	9.83e-03	1.92e-02
519	75	8km	3	max	3.03e-03	1.68e-03	1.09e-02	2.04e-02	2.79e-02
520	75	8km	4	mean	2.01e-03	1.23e-03	6.77e-03	1.27e-02	2.07e-02
521	75	8km	4	med	2.02e-03	1.26e-03	6.80e-03	1.30e-02	2.48e-02
522	75	8km	4	max	4.09e-03	2.39e-03	1.42e-02	2.64e-02	3.63e-02
523	75	8km	5	mean	1.75e-03	1.05e-03	5.99e-03	1.12e-02	1.84e-02
524	75	8km	5	med	1.76e-03	1.08e-03	6.01e-03	1.14e-02	2.20e-02
525	75	8km	5	max	3.56e-03	2.03e-03	1.25e-02	2.34e-02	3.22e-02
526	75	12km	1	mean	3.12e-06	1.77e-06	1.12e-05	2.34e-05	3.32e-05
527	75	12km	1	med	3.14e-06	1.83e-06	1.10e-05	2.34e-05	3.87e-05
528	75	12km	1	max	6.34e-06	3.42e-06	2.32e-05	5.02e-05	6.52e-05
529	75	12km	2	mean	3.22e-04	2.31e-04	8.98e-04	1.53e-03	2.07e-03
530	75	12km	2	med	3.29e-04	2.38e-04	9.20e-04	1.60e-03	2.40e-03
531	75	12km	2	max	6.48e-04	4.49e-04	1.90e-03	3.24e-03	4.19e-03
532	75	12km	3	mean	9.01e-04	5.81e-04	2.87e-03	4.95e-03	7.13e-03
533	75	12km	3	med	9.19e-04	6.00e-04	2.90e-03	5.20e-03	8.27e-03
534	75	12km	3	max	1.82e-03	1.12e-03	6.05e-03	1.08e-02	1.44e-02
535	75	12km	4	mean	1.23e-03	8.28e-04	3.75e-03	6.40e-03	9.23e-03
536	75	12km	4	med	1.25e-03	8.62e-04	3.80e-03	6.78e-03	1.07e-02
537	75	12km	4	max	2.47e-03	1.61e-03	7.94e-03	1.40e-02	1.86e-02
538	75	12km	5	mean	1.07e-03	7.01e-04	3.31e-03	5.68e-03	8.19e-03
539	75	12km	5	med	1.09e-03	7.30e-04	3.34e-03	6.00e-03	9.51e-03
540	75	12km	5	max	2.15e-03	1.37e-03	7.00e-03	1.24e-02	1.65e-02