



Working Group on Design and Safety Analysis

Phase 4 Report

Mechanistic Source Term

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EXECUTIVE SUMMARY

There continues to be sustained global interest in small modular reactors (SMRs), which have the potential to play an important role in globally sustainable energy development as part of an energy mix. In particular, SMRs may enhance energy availability and security of supply in countries expanding their nuclear energy programs and those embarking on a nuclear energy program for the first time.

As the interest in SMRs continues to grow, so does the importance of international collaboration. Given that its main purpose is to bring together experienced regulatory bodies to identify and address key SMR-related challenges, the SMR Regulators' Forum (SMR RF) has an increasingly important role to play in making such collaboration possible.

The SMR RF was formed in 2014 as a regulator-to-regulator entity to consider key issues that could emerge in future SMR regulatory discussions and propose common positions regarding the way in which these could be addressed. The Forum's work is expected to help enhance safety as well as efficiency in SMR regulation, including licensing, and to enable regulatory bodies to inform changes, if necessary, to their requirements and regulatory practices. Since then, the Forum has had four phases of work. For more details about the Forum, please visit: <https://gnssn.iaea.org/regnet/SMR-Forum/Pages/default.aspx>

This report has been produced by the Design and Safety Analysis (DSA) Working Group (WG) of the SMR RF during its working Phase 4 (2024 to 2026). It explores the application of mechanistic source term (MST) methodologies in the safety analysis of SMRs, particularly non-light water reactor (non-LWR) designs. Unlike traditional conservative approaches, MST uses realistic, scenario-specific modelling to assess the release and transport of radionuclides during nuclear accidents. This MST approach enables more accurate assessments tailored to novel reactor designs, some of which do not rely on traditional containment structures or assume core melt scenarios.

This report was developed based on information, insights, and experience gained from the regulatory activities of the SMR RF Members. While it is generally aligned with existing IAEA documents, deviations may occur in some cases. This report aims to provide useful information to regulatory bodies and industry involved in the development, deployment and oversight of SMRs.

The text presents 'common positions', that is, statements relevant to the topic under discussion that are unanimously agreed upon by the Forum Members - on issues such as MST applicability, methodology and process, including identification of initiating events and phenomena, calculation, as well as sensitivity and uncertainty analysis. It is important to note that these common positions reflect the views of the Forum's experts and do not represent a formal consensus of all IAEA Member States.

The common positions developed by the DSA WG are outlined below and include recommendations for potential updates to the IAEA safety standards.

COMMON POSITIONS FOR THIS REPORT

Common Position 1

In principle, MST methodologies could be applied to all accident categories and for all technology types.

Common Position 2

Regardless of whether the MST methodology is applied, adequate conservatism should be incorporated when determining the radiological consequences.

Common Position 3

When MST methodologies are applied, a demonstration of adequate understanding of the accident progression, the relevant phenomena and their modelling, methods, and assumptions should be provided.

Common Position 4

The process for determining the MST should be followed irrespective of the reactor technologies, and should include the following steps, not necessarily in that order: Scenario Identification, Phenomena Analysis, Source Term Calculation, and Sensitivity and Uncertainty Analysis.

Common Position 5

Initiating event/sequence identification should be developed on the basis of the whole reactor facility, including all systems where radionuclides reside.

Common Position 6

In the context of MST determination, the process for the identification and categorization of initiating events/sequences may differ from traditional approaches. The approach for identification and categorization of initiating events/sequences should be justified, particularly for novel designs.

Common Position 7

Existing initiating events/sequences lists should not be the main source of initiating event/sequence identification for novel designs, but rather should be used as a supplement.

Common Position 8

The vendor/licensee should demonstrate an adequate understanding of accident progression and phenomenology to the regulatory body. The vendor/licensee should make available any relevant documentation, including, but not limited to, the Phenomena Identification and Ranking Tables (PIRTs) when available.

Common Position 9

The licensee/vendor should demonstrate that the codes and models used in determining the MST are adequately verified and validated.

Common Position 10

The MST calculations should be such that the phenomena influencing fission product release are not masked. For example, in some cases, if the calculations are too conservative, the fission product behaviour resulting in the release may not be captured properly or in sufficient detail.

Common Position 11

MST uncertainty analysis and/or sensitivity studies should be performed, documented, and made available to the regulatory body.

Recommendations for IAEA

Potential changes to SSG-2, SSG-3 and SSG-4.

1. INTRODUCTION

The use of postulated accidental release of radioactive materials and consequent radiological doses has long been deeply embedded in the regulatory policy and practices in the siting and licensing of nuclear reactors and protection of public health. The IAEA *Nuclear Safety and Security Glossary* [1] define source term as “the amount and isotopic composition of radioactive material released (or postulated to be released) from a facility”. In the early days of commercial nuclear power, licensing events were identified on a case-by-case basis, relying on the collective engineering judgment of designers and the regulatory staff. Traditionally, the source term used in light water reactor (LWR) siting evaluations was based on an in-vessel core melt accident, in which a large portion of the radionuclide inventory is released early into the environment. This entailed the use of conservative, non-mechanistic methods to yield conservative dose estimates and demonstrate compliance with regulatory requirements.

Because the concept of core melt does not apply to some SMRs, in the recent non-light water reactor (non-LWR) pre-application activities, the designers propose an approach to source term selection using mechanistic scenario-specific source terms derived from specific initiating events defined for the plant, using phenomenological models of fission product transport. For LWRs, significant progress has been made through the years in understanding reactor accident behavior, including fission product release and transport. This enhanced technical understanding enables more detailed assessments of mechanistic source term (MST), which is appropriate for advanced reactors. However, MSTs developed for light-water reactors (LWRs) are reactor-specific and are not applicable to non-LWR small modular reactors (SMRs). The global trend for advanced non-LWRs has been to develop MSTs based on selected design basis events, supported by design-specific analyses and available experimental data on fuel behavior and core response under off-normal conditions. For example, the HTR-10 in China applied an MST approach in which severe core damage was not assumed a priori for siting purposes. Instead, radiological releases were calculated explicitly for specific accident sequences, and the source term was based on those events resulting in the highest radionuclide releases from the fuel.

1.1. SMR REGULATORS' FORUM PREVIOUS WORK IN THE AREA OF DESIGN AND SAFETY ANALYSIS

The SMR Regulators' Forum (SMR RF) began with a Pilot phase (2015 to 2017), during which important topics in relation to the design and safety analysis were covered [2], including:

- Use of a Graded Approach;
- The implications of SMR characteristics on Defence in Depth;
- Documenting considerations in implementing flexible Emergency Planning Zones.

In Phase 2 (starting in 2018), a Design and Safety Analysis Working Group (DSA WG) was established, and it has continued until Phase 4 (2024-2026), examining several critical topics:

- Multi-unit aspects [3] and [4]
- Uncertainties arising from the use of passive systems in SMR [3]
- Design Extension Conditions (DEC), previously known as beyond design basis accidents (BDBAs) [3]
- Containment systems [5]
- Safety, security and safeguards from a regulatory perspective [6]
- Safety, security and safeguards in remote locations, currently under development.

The precursor for this report on MST is the work undertaken by the DSA WG in Phase 3, examining the concept of functional containment in different Forum's Members regulatory frameworks reflected in report titled *Containment Systems* [5] . The concept of functional containment is important, because some advanced non-LWRs claim that a leak-tight and pressure retaining containment structure is not relied upon to restrict the consequences of accidents. This claim is based on inherent and passive safety, such as accident tolerant fuels and other protection features, which are intended to reduce the reliance on the structure and its associated systems to provide the containment function. These designs may propose alternative measures to limit radionuclide releases to the environment, for example, retention of radionuclides at their source in the fuel rather than assuming extensive fuel particle or fuel element failures and subsequent reliance upon other barriers (the reactor coolant system and containment structure) to ensure that dose at the site boundary meets regulatory limits resulting from postulated accidents. For such designs, in the absence of a pressure-retaining containment structure, the United States Nuclear Regulatory Commission (U.S. NRC) uses the term 'functional containment' [7] .

The concept of MST is closely intertwined with functional containment. Regardless of the method used to achieve functional containment, all reactor designs must prioritize minimizing the release of radioactive materials following an accident. It is crucial to keep this release as minimal as reasonably possible and within regulatory limits.

The Phase 3 discussions within the DSA WG on this matter were reflected in the relevant SMR RF Report [5] .

In phase 4 of the SMR RF, the DSA WG is set to:

- investigate Safety, Security and Safeguards considerations of an SMR in Remote Locations;
- develop common regulatory bodies' positions pertaining to MST.

This report focuses on the application of MST methodologies for SMRs.

1.2. IAEA CONTEXT

The IAEA General Safety Requirement *Safety Assessment for Facilities and Activities* [8] , requirement 6, states that "[t]he possible radiation risks associated with the facility or activity shall be identified and assessed".

The IAEA Specific Safety Guide *Deterministic Safety Analysis for Nuclear Power Plants* [9] states in section 2.16 that “[a]n essential component of deterministic safety analysis is the determination of source terms for releases of radioactive material as a key factor for prediction of dispersion of such material in the environment and ultimately of radiation doses to plant staff and the public as well as the radiological impact on the environment”. Section 2.17 states that:

“[t]o evaluate the source term from a nuclear installation, it is necessary to identify the sources of radiation; to determine the inventories of radionuclides that are produced and to know the mechanisms by which radioactive material can travel from the source through the installation and be released to the environment. Under accident conditions, source term evaluation requires simulation codes capable of predicting fission product release from fuel elements, transport through the primary system and containment or spent fuel pool building, the related chemistry affecting this transport and the form in which the radioactive material would be released”.

In addition, the IAEA Specific Safety Guide *Development and Application of a Level 2 Probabilistic Safety Analysis for Nuclear Power Plants* [10] has detailed recommendations on how to conduct a source term analysis for severe accidents.

The IAEA Safety Report *Applicability of IAEA Safety Standards to non-Water Cooled Reactors and Small Modular Reactors* [11] identifies insufficient guidance in SSG-2 [9] and SSG-4 [10] for the determination of source terms for novel technologies such as non-water cooled SMRs. This publication represents regulatory common positions on MST methodologies and their applicability to novel reactor designs that may help inform future work in this area.

2. BACKGROUND

Radiological consequence assessment is crucial for demonstrating compliance with public dose acceptance criteria - or alternative criteria in the case of design extension conditions (DECs) or beyond design basis accidents (BDBAs) - as required/expected by the regulatory bodies. An evaluation of the source term from a facility forms an important input into the radiological consequence assessment.

Specifically in 10 CFR 50.2 [12], the United States Nuclear Regulatory Commission (U.S. NRC) defines source term as “the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release”.

This definition of the source term should not be confused with the MST definition. The term MST is more specific and often used in the context of accident analysis or radioactive material transport. Currently, there is limited guidance that provides a definition of MST that is applicable to both LWR and non-LWR technologies. The U.S. NRC has previously described a MST in SECY-03-0047[7] as the result of an analysis of fission product release resulting from the design-specific accident scenarios and accident progression being evaluated. It is developed using best-estimate phenomenological models with uncertainty quantification of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environment. It should be emphasized that such an approach needs to be supported by research and testing to fully understand and develop a technically sound MST.

Based on this definition, for the purposes of this report, the DSA WG considers an MST methodology to have the following characteristics:

- it is accident scenario and facility specific;
- it applies best-estimate models and assumptions to realistically model the facility and relevant phenomena;
- mechanistically, it is based on underlying physical mechanisms, determines the release of all relevant radionuclides from its normal location and transport to the point of release to the environment;
- it provides a representation of the accident and release with time evolution.

In addition to radiological hazards, some novel designs (especially non-LWRs) may introduce conventional (non-radiological) hazards resulting from initiating events/sequences. For example, lead may be released into the environment. Potential chemical interactions of the released substances with air and/or the formation of compounds that may be harmful in ways beyond radioactivity should be accounted for in safety analyses of the novel designs. For the purposes of this report, MST only focuses on radiological aspects resulting from an initiating event/sequence.

2.1. HISTORICAL PERSPECTIVES ON MECHANISTIC SOURCE TERM

The Three Mile Island (TMI) accident, which occurred in 1979 in Pennsylvania, USA, provides a significant historical perspective on MST analysis. During this event, a series of equipment malfunctions and human errors led to a partial meltdown of the reactor core. As a result, significant quantities of radioactive gases, including xenon-133, and krypton-85, and volatile iodine-131 were released into the containment building, with only very small amounts of volatile radionuclides escaping into the surrounding environment due to a breach in the containment system [14]

The MST analysis conducted after the TMI accident [15] played a crucial role in understanding the sequence of events and quantifying the extent of radioactive releases. These analyses provided key insights into containment performance and informed the evaluation of potential radiological consequences. The lessons learned contributed to subsequent regulatory reforms and safety enhancements in the nuclear industry, including improvements in accident analysis methodologies and operator training programs.

The Fukushima Daiichi nuclear accident in 2011, following a massive earthquake and tsunami in Japan, offers another compelling case study for MST analysis. The severe accident at the Fukushima Daiichi Nuclear Power Plant (NPP) resulted in core meltdowns, hydrogen explosions, and significant releases of radioactive materials, including both volatile and particulate radionuclides, into the atmosphere and surrounding environment.

MST analysis conducted in the aftermath of the Fukushima accident [16] provided valuable insights into the behavior of radioactive materials under severe accident conditions, helping to quantify releases and evaluate the performance of containment structures. The findings from this analysis prompted global efforts to enhance nuclear safety standards, emergency preparedness, and accident mitigation measures.

These case studies highlight the importance of the MST analysis in understanding the dynamics of nuclear accidents, evaluating their consequences, and informing regulatory decision-making and safety improvements in the nuclear industry.

Since LWRs form the large majority of existing reactors, there is vast knowledge and understanding of accident progression in LWRs. As a result, conservative source term methodologies are commonly applied for accidents that do not involve fuel melting. In contrast, advanced nuclear technologies include a wide variety of reactor designs, with accident progression governed by complex physiochemical phenomenology. For many of these technologies, the traditional LWR-based demarcation between design basis and beyond design basis (including severe) accidents cannot be applied. Consequently, in addition to the severe accident analysis of LWRs, MST methodologies are particularly valuable for non-LWR reactors, as they provide a mechanistic framework applicable across a broad spectrum of accident scenarios.

Non-LWRs

In the US, operating LWRs were licensed based on TID-14844 [17], a document published in 1962 by the United States Atomic Energy Commission. This document assumes a 'substantial meltdown of the core', leading to the release of 100% of noble gases, 50% of

halogens, and 1% of solids into the containment, representing roughly 15% of the total activity in the core. Once within the containment, some attenuation of the released activity is permitted, depending on the available systems. However, this source term is considered conservative, non-mechanistic, and not reflective of realistic system behavior or experimental data. Additionally, it is not applicable to non-LWRs.

Regulatory Guide 1.183 [18] [18] provides guidance for licensees on how to build an alternative (mechanistic) source term to that provided in NUREG-1465 [19] and TID-14844 [17] for LWRs. For non-LWRs, significantly different physics and chemistry present within non-LWRs compared to operating LWRs necessitates different guidance. A guideline for a technology-inclusive, risk-informed, performance-based approach for the selection of event scenarios, safety systems and correlating risk-informed treatment of the system is provided in the Licensing Modernization Project (LMP), which was endorsed by the NRC in SECY-19-0177 [20] and Regulatory Guide 1.233 [21] .

2.2. REGULATORY FRAMEWORK AND REQUIREMENTS RELATED TO MECHANISTIC SOURCE TERM

The DSA WG members were tasked with providing insights into their regulatory frameworks and requirements related to MST through the questionnaire given in Appendix A.

The questionnaire on MST is designed to gather insights regarding regulatory frameworks for assessing radioactive material release in nuclear accidents. It focuses on how MSTs are addressed in regulatory requirements and their applicability to Generation IV (Gen-IV) SMRs. Key areas of focus included:

- **Regulatory requirements for MST**- whether existing regulatory requirements address these terms directly or if broader safety regulations could be applied.
- **Identification and categorization of postulated initiating events (PIEs)** - examining how events like Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs), and Beyond Design Basis Accidents (BDBAs) are handled in current regulatory requirements, and whether these are adaptable to new reactor designs.
- **Radionuclide lists** - assessing if countries use fixed lists of radionuclides for safety evaluations, and whether these lists are relevant to SMRs.
- **Quantitative safety goals** - looking at how safety goals for radionuclide releases, such as I-131 and Cs-137, are defined and if they need updating for newer reactors.

Ultimately, the questionnaire aimed to identify gaps or needs for updating regulatory frameworks to ensure the safe design and operation of next-generation nuclear reactors.

3. COMMON POSITIONS

3.1. MECHANISTIC SOURCE TERM APPLICABILITY

In general, regulatory bodies do not prescribe a specific methodology for source term evaluation. Practices in Member States range from more simplistic representations of phenomena to sophisticated best-estimate methods representing complex scenarios. The choice of methods depends on the level of understanding of phenomena, understanding of accident progression, likelihood of an event or sequence and expected conservatism to be adopted. Generally, regulatory bodies expect that a conservative approach should be taken for source term evaluation of design basis accidents whereas a more mechanistic and best-estimate approach is expected for severe accident scenarios.

Common Position 1

In principle, MST methodologies could be applied to all accident categories and for all technology types.

Common Position 2

Regardless of whether the MST methodology is applied, adequate conservatism should be incorporated when determining the radiological consequences.

3.2. MECHANISTIC SOURCE TERM METHODOLOGY

Even though most countries do not define MST, some regulatory bodies' current regulatory requirements or expectations include a source term that in essence represents the 'mechanistic' source term. Where these methodologies are applied, regulatory bodies generally have the following expectations:

- The modelling methods used to perform source term analysis are described and justified.
- The radionuclide grouping scheme used for the source term analysis is consistent with current state-of-the-art practices or must be justified.
- All the models used, and assumptions made to obtain source terms are justified and documented.
- The computer codes used for estimating facility-specific source terms for selected accident sequences are verified and validated.
- If there are cases where facility-specific computer code calculations were not performed, or phenomena are not fully represented, the method by which source terms have been estimated is described and justified.

Common Position 3

When MST methodologies are applied, a demonstration of adequate understanding of the accident progression, the relevant phenomena and their modelling, methods, and assumptions should be provided.

3.3. MECHANISTIC SOURCE TERM PROCESS

The process for determining the MST involves a systematic analysis of the sequence of events and physicochemical phenomena occurring during a nuclear accident. While the specific steps may vary depending on the reactor type and regulatory requirements, the general process typically includes the following key stages:

- **Scenario Identification:** the first step is to identify the initiating events or sequences that could lead to a radioactive release from all facilities within the NPP site.
- **Phenomena Analysis:** once the initiating events are identified, a detailed analysis of the phenomena involved in each scenario is conducted. This includes, but is not limited to, assessing core degradation mechanisms, fuel and coolant behaviour, fission product release pathways, and containment response.
- **Source Term Calculation:** based on the understanding of the accident sequence and physicochemical phenomena, calculations are performed to estimate the release of radioactive materials from their normal location of confinement to the environment. This involves modelling the transport and behaviour of radioactive isotopes under various conditions.
- **Sensitivity and Uncertainty Analysis:** sensitivity and uncertainty analyses are conducted to evaluate the impact of uncertainties in input parameters and assumptions on the source term calculations. This helps assess the robustness of the results and identify areas where additional data or analysis may be needed.

The determination of the MST in nuclear accidents involves a multifaceted process that integrates phenomena analysis and source term calculation. This process involves understanding the intricate physical and chemical mechanisms governing the release of radioactive materials during an accident. Investigating various scenarios and release pathways is crucial, considering factors such as fuel behaviour, coolant interactions, and barrier integrity.

Common Position 4

The process for determining the MST should be followed irrespective of the reactor technologies and should include the following steps, not necessarily in that order: Scenario Identification, Phenomena Analysis, Source Term Calculation, and Sensitivity and Uncertainty Analysis.

3.3.1. INITIATING EVENT/SEQUENCE IDENTIFICATION

An important part of the reactor design process is the identification of events that could challenge fundamental safety functions and layers of defence against the release of radioactive materials. Regulatory bodies expect that the designer should systematically derive initiating events/sequences, demonstrate an understanding of the progression of those initiating events/sequences, and identify protective safety measures to maintain the fundamental safety functions and defence in depth.

Regulatory bodies typically expect multiple methods to be applied when performing initiating event/sequence identification. Hazard and Operability Analysis (HAZOP) and Failure Mode and Effect Analysis (FMEA) are amongst the most common methodologies employed in the design phase. While existing lists of initiating events/sequences based on other reactor designs are often used to assist in initiating events/sequences identification, this will likely be challenging for many of the novel designs or technologies which are relatively less well understood. Preconceptions based on previous designs may also be detrimental to the initiating events/sequences identification process. Applicants should address the uncertainties inherent in scenario identification for novel designs, arising from gaps in the knowledge base, by conducting thorough, unbiased searches for potential initiating events/sequences, without any preconceived notions that could constrain the creative process.

The design process typically starts with deterministic identification of initiating event/sequence. During the design development, the usefulness of probabilistic safety analysis (PSA) may be limited by the insufficient knowledge of the likelihood of initiating events/sequences due to lack of operating experience of novel designs.

In the context of MST, initiating event/sequence identification and categorization is important for the level of conservatism employed in the analysis.

Since traditional categorization of reactor initiating events/sequences is largely based upon the resulting fuel state (e.g. AOOs, DBAs, DEC-A and DEC-B), challenges may exist in assigning initiating events into traditional categories for reactors such as molten salt reactors. For example, in molten salt reactors, the fuel is dissolved and therefore the concept of core melt does not apply. In such cases, it is recognized by regulatory bodies that novel approaches to categorization of initiating events/sequences may be required. There have been many attempts in the open literature to categorize initiating events/sequences.

In some reactor types (for example, molten salt reactors, liquid metal fast reactors), radionuclides contributing to the overall source term may reside in auxiliary systems.

Common Position 5

Initiating event/sequence identification should be developed on the basis of the whole reactor facility, including all systems where radionuclides reside.

Common Position 6

In the context of MST determination, the process for the identification and categorization of initiating events/sequences may differ from traditional approaches. The approach for identification and categorisation of initiating events/sequences should be justified, particularly for novel designs.

Common Position 7

Existing initiating events/sequences lists should not be the main source of initiating event/sequence identification for novel designs but should rather be used as a supplement.

3.3.2. PHENOMENA ANALYSIS

Accident progression in nuclear reactors involves many physical and chemical phenomena that significantly impact the potential magnitude and composition of radiological release. For LWRs, phenomena in accidents in which fuel failure is prevented are well researched, understood, and implemented in computer models. This knowledge can be transferred and implemented for LWR SMRs. However, for severe accidents and non-LWR SMRs, accident sequences and phenomenology are relatively less well understood. Large uncertainties remain in complex accident sequences. This is partly due to insufficient research, or inherent limitations in knowledge due to the complex nature of severe accident progression.

Regulatory bodies expect that all appropriate phenomena that influence accidents and radionuclide transport progression should be understood. Gaining an understanding of accident progression and release may involve:

- modelling the behaviour of fission products including their release from fuel, transport in the reactor coolant, and potential interactions with reactor structures, systems and components;
- accounting for aerosol formation and transport as they play a significant role in the dispersion of radioactive materials;
- considering a wide range of physical and chemical processes.

By integrating phenomena analysis into MST calculations, vendors can gain valuable insights into the potential consequences of nuclear accidents. This comprehensive understanding informs regulatory decision-making, emergency preparedness efforts, and safety enhancements in the nuclear industry.

Common Position 8

The vendor/licensee should demonstrate an adequate understanding of accident progression and phenomenology to the regulatory body. The vendor/licensee should make available any relevant documentation, including, but not limited to, the Phenomena Identification and Ranking Tables (PIRTs), when available.

3.3.3. MST CALCULATION

Computer codes used to model all phenomena of the accident progression for which the MST is being calculated need to be properly qualified through extensive verification and validation. This is done through the research, development, and testing programmes. This information should be made available to the regulatory bodies in sufficient detail to facilitate regulatory review.

MST may not be applicable in circumstances when computer codes and models cannot be properly validated.

Common Position 9

The licensee/vendor should demonstrate that the codes and models used in determining the MST are adequately verified and validated.

Common Position 10

The MST calculations should be such that the phenomena influencing fission product release are not masked. For example, in some cases, if the calculations are too conservative, the fission product behaviour resulting in the release may not be captured properly or in sufficient detail.

3.3.4. SENSITIVITY AND UNCERTAINTY ANALYSIS

Due to the limited knowledge of accident progression and phenomena in non-LWR designs, larger uncertainties typically exist in the modelling of phenomena leading to their environmental releases compared to those of LWRs designs. Quantification of the uncertainties provides insights into the accuracy of the code predictions and is important to regulatory decision-making.

Sensitivity studies may provide useful insights and may be used to derive the most challenging conditions in demonstrating compliance with relevant limits or may be used to demonstrate the absence of cliff-edge effects.

Common Position 11

MST uncertainty analysis and/or sensitivity studies should be performed, documented, and made available to the regulatory body.

3.3.5. IAEA GUIDANCE FOR NOVEL NON-LW SMRS

When reviewing the accident identification and classification provided by vendors/utilities, the regulatory bodies should consult the IAEA Safety Report 123 *Applicability of IAEA Safety Standards to Non-Water Cooled Reactors and Small Modular Reactor* [11] as a generic, but useful reference, which examines multiple technologies.

More specifically, for High Temperature Gas Reactors (HTGRs), the regulatory bodies should consult the IAEA safety report *Accident Analysis for Nuclear Power Plants with Modular High Temperature Gas Cooled Reactors* [22] .

When reviewing the source term calculations provided by vendors/utilities, the regulatory bodies are advised to consult relevant IAEA publications. For HTGRs these include, but are not limited to:

- *Fuel Performance and Fission Product Behavior in Gas-Cooled Reactor* [23] , with specific examples of source term calculations for the HTR-MODUL. This document provides fuel performance data, fission product behavior, and numerical codes to consider in source term predictions.

- *Advances in High Temperature Gas Cooled Reactor Fuel Technology* [24] . Chapter 10 of this document provides benchmarking of fission products release behavior models under accident conditions, while Chapter 12.5 provides accident source term analysis.
- *Safety Related Design and Economic Aspects of HTGRs* [25] , which contains a paper from China on relevant safety issues in designing the HTR-10 reactor. Chapter 6 of this paper contains a discussion on the mechanistic approach adopted for HTR-10.

For Liquid Metal Fast Reactors, the regulatory bodies are advised to consult the IAEA publication *Modelling and Simulation of the Source Term for a Sodium Cooled Fast Reactor Under Hypothetical Severe Accident Conditions* [26] , as well as the recently published *Analysis and Modelling of Severe Accidents for Liquid Metal Fast Reactor* [27] , which includes considerations of the source term.

In addition, the IAEA has several publications under development addressing the safety aspects of non-water-cooled technologies including *Considerations on the Safety of HTGRs*, *Considerations on the Safety of Liquid Metal Fast Reactors* and *Considerations on the Safety of Molten Salt Reactors*.

4. CONCLUSION

This report explores the application of MST methodologies in the safety analysis of SMRs, particularly non-LWR designs. Unlike traditional conservative approaches, MST uses realistic, scenario-specific modelling to assess the release of radionuclides during nuclear accidents. The MST approach enables more accurate assessments tailored to novel reactor designs, many of which do not rely on traditional containment structures or assume core melt scenarios.

The report presents 11 common positions on issues such as: MST applicability, methodology, process, scenario identification, calculation, and sensitivity and uncertainty analysis. These will be useful to regulatory bodies and industry involved in the development, deployment and oversight of SMRs. In addition, they should be helpful in future updates to SSG-2 [9] , SSG-4 [10] and other relevant IAEA safety standards.

**APPENDIX A. QUESTIONNAIRE ON MECHANISTIC SOURCE TERM REFLECTING
INFORMATION AS OF 2024**

1. Do you have regulatory requirements pertaining to mechanistic source terms?
2. If not, what existing regulatory requirements could be used in your reviews of mechanistic source terms?
3. Do you have requirements for identification and categorization of postulated initiating events (covering AOOs, DBAs and DEC/BDBAs)?
4. Would your existing requirements for identification and categorization of PIEs - postulated initiating events (covering AOOs, DBAs and DEC/BDBAs) be applicable to reviews of SMRs including Gen-IV SMRs? If not, why and how would you go about it? Please describe, in your professional opinion (does not have to represent the organizational 'consensus'), what thinking/method would you use to assess PIEs?
5. Does your country use a source term only covering specific radionuclides (source term 'list')? If yes, is the list adequate and relevant to SMRs including Gen-IV SMRs?
6. In your country, how are the quantitative safety goals on releases (typically I-131 and Cs-137 are the radionuclides used in quantitative safety goals) defined? Would these quantitative safety goals be applicable to SMRs including Gen-IV SMRs? If not, what would be the regulatory process for an exemption?

APPENDIX B - FORUM'S MEMBERS RESPONSES TO THE QUESTIONNAIRE

APPENDIX B.1 - CANADA

Canadian Nuclear Safety Commission (CNSC) response to the Questionnaire on Mechanistic Source Term

1. Do you have regulatory requirements pertaining to mechanistic source terms?

The term “mechanistic source term” is not used in our regulatory framework.

If not, what existing regulatory requirements could be used in your reviews of mechanistic source terms?

While the term “mechanistic source term” is not explicitly defined in CNSC regulatory requirements, several existing requirements provide a clear basis for reviewing source term evaluations for all plant states, as defined in REGDOC-2.5.2 [28]. Key requirements include:

REGDOC-2.4.1 – Deterministic Safety Analysis (DSA) [29]:

Section 4.2: Requires identification and analysis of all postulated initiating events (PIEs) for anticipated operational occurrences (AOOs), design-basis accidents (DBAs), and beyond-design-basis accidents (BDBAs), providing a framework for scenario-based source term evaluation.

Section 4.3: Specifies that analyses must predict radiological releases from fuel and their transport through barriers, including containment performance, to demonstrate compliance with regulatory dose criteria.

Section 4.5: Mandates that assumptions and analysis methods be justified and validated using experimental, operational, or benchmark data.

Section 3.2: Specifies that one of the objectives of DSA is to “assist in demonstrating that safety goals - which may be established to limit the risks posed by the NPP - are met” and to “predict source term and doses during severe accidents”.

Do you have requirements for identification and categorization of postulated initiating events (covering AOOs, DBAs and DEC/BDBAs)?

Yes. These requirements are listed in Section 8.2.3 ‘Classification of events’ of REGDOC-2.4.1 [29] as follows:

“The identified events shall be classified, based on the results of PSA and engineering judgment, into the following three classes of events:

- anticipated operational occurrences (AOOs), which include all events with frequencies of occurrence equal to or greater than 10^{-2} per reactor year.

- design-basis accidents (DBAs), which include all events with frequencies of occurrence equal to or greater than 10^{-5} per reactor year but less than 10^{-2} per reactor year. This class of events also includes any events that are used as a design basis for a safety system, regardless of whether the estimated frequencies are less than 10^{-5} per reactor year.
- beyond-design-basis accidents (BDBAs), which include events with frequencies of occurrence less than 10^{-5} per reactor year.

Events with a frequency near the threshold between two classes of events, or with substantial uncertainty over the predicted event frequency, should be classified into a higher frequency class.

Credible common cause events shall also be classified within the AOO, DBA and BDBA classes.”

Various other requirements pertaining to AOOs, DBAs and BDBAs are specified in the Refs [28] and [29] , as well as many other Canadian regulatory documents and standards.

2. **Would your existing requirements for identification and categorization of PIEs - postulated initiating events (covering AOOs, DBAs and DEC/BDBAs) be applicable to reviews of SMRs including Gen-IV SMRs? If not, why and how would you go about it? Please describe, in your professional opinion (does not have to represent the organizational ‘consensus’), what thinking/method would you use to assess PIEs?**

Our existing requirements for identification and categorization of PIEs would be applicable for reviews of SMRs including Gen-IV SMRs. However, when classifying the events based on the results of PSA, uncertainties must be carefully considered.

3. **Does your country use a source term only covering specific radionuclides (source term ‘list’)? If yes, is the list adequate and relevant to SMRs including Gen-IV SMRs?**

No. Canada does not have such list.

4. **In your country, how are the quantitative safety goals on releases (typically I-131 and Cs-137 are the radionuclides used in quantitative safety goals) defined? Would these quantitative safety goals be applicable to SMRs including Gen-IV SMRs? If not, what would be the regulatory process for an exemption?**

Section 2.2.2 ‘Safety goals’ of REGDOC-2.5.2 [28] defines:

“Qualitative safety goals

A limit is placed on the societal risks posed by reactor facility operation. For this purpose, the following 2 qualitative safety goals have been established:

Individual members of the public shall be provided a level of protection from the consequences of reactor facility operation, such that there is no significant additional risk to the life and health of individuals.

Societal risks to life and health from reactor facility operation shall be comparable to or less than the risks of generating electricity by viable competing technologies, and shall not significantly add to other societal risks.

Quantitative application of the safety goals

For practical application, the following 3 quantitative safety goals have been established so as to achieve the intent of the qualitative safety goals:

- core damage frequency
- small release frequency
- large release frequency

A core damage accident results from a postulated initiating event (PIE) followed by the failure of 1 or more safety system(s) or safety support system(s). Core damage frequency is a measure of the plant's accident prevention capabilities.

Small release frequency and large release frequency are measures of the plant's accident mitigation capabilities. They also represent measures of risk to society and to the environment due to the operation of reactor facilities.

Core damage frequency

The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 10^{-5} per reactor year.

Small release frequency

The sum of frequencies of all event sequences that can lead to any release to the environment that requires temporary evacuation of the local population or a release to the environment of more than 10^{15} becquerels of iodine-131 shall be less than 10^{-5} per reactor year.

Large release frequency

The sum of frequencies of all event sequences that can lead to any release to the environment that requires long-term relocation of the population or a release to the environment of more than 10^{14} becquerels of cesium-137 shall be less than 10^{-6} per reactor year."

APPENDIX B.2 - FINLAND

Radiation and Nuclear Safety Authority (STUK) Response to Questionnaire on Mechanistic Source Term

NOTE: In Finnish legislation, the term 'accident' refers to postulated accidents, design extension conditions (which differs from IAEA definition of DEC¹) and severe accidents (core melt scenarios).

1. Do you have regulatory requirements pertaining to mechanistic source terms?

This term is not used.

2. If not, what existing regulatory requirements could be used in your reviews of mechanistic source terms?

We have a set of requirements for release and radiation dose analyses that could be applied to MST analysis, as presented hereinafter.

- Section 24 'Radiation measurements and monitoring of releases of radioactive substances at a nuclear facility and estimation of radiation doses to the public and workers' of STUK Y/1/2018 regulation [30] :

"2. The radiation doses to the workers and the public in the surroundings caused by the operation or decommissioning of a nuclear facility shall be measured or otherwise estimated with due consideration given to external and internal radiation exposure.

3. As concerns the public's radiation doses, the radiation dose for an individual representing the most exposed group shall be defined. In the definition of radiation exposure, the significant migration routes of radioactive substances shall be taken into account."

- Chapters 3 'Events to be analysed' and 5 'Release and radiation dose analyses' of YVL B.3 guide:

"301. Analyses pertaining to the plant's behaviour as well as releases of radioactive substances and radiation doses caused by the releases shall cover the nuclear power plant's normal operational states, anticipated operational occurrences, postulated accidents, design extension conditions and severe reactor accidents. Examples of the events to be analysed are given in GSR Part 4 (Rev. 1) [8] and SSG-2 (Rev. 1)[9]

...

5.1 Events to be analysed

¹ According to the IAEA Nuclear Safety and Security Glossary [1] design extension conditions are "Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits."

501. Release and radiation dose analyses shall be performed on those transient and accident cases required in para 301 which are limiting as regards the release of radioactive substances and radiation doses. Selection of the limiting cases shall be justified.

502. The analyses referred to in para 501 shall be supplemented with an analysis of the containment's retention capability, in which the source term into the containment is calculated according to the maximum number of failed fuel rods (10%) allowed in a class 2 postulated accident as required in para 417 of Guide YVL B.4.

...

5.3 Assumptions for release and dose related analyses

5.3.1 General assumptions

506. In analysing releases, the same assumptions shall be used to describe the plant as are used in the analyses in chapter 4.3.

507. At least the same amount of radioactive substances shall be assumed in the primary coolant at the beginning of an event as is set as the maximum limit in the Technical Specifications of the plant.

508. The number of leaking fuel rods before an accident shall be chosen in conformity with para 507.

509. The increasing release of fission products due to the pressure difference change between the failed fuel and coolant shall be taken into account in evaluating the concentration of radioactive substances in the primary coolant. The increase in concentration and its dependence on time shall be justified.

510. The effect of cooling water that enters the failed fuel rods on the release of radioactive substances shall be taken into account in the analyses.

511. The distribution of radioactive substances into gaseous and liquid phases of the leaking substance shall be justified.

512. A fraction of the iodine mixed with the steam shall be assumed to be gaseous. The distribution of iodine into gaseous and aerosol phases shall be justified.

513. The halogens released into airspace shall be assumed to be partly bound to inorganic and partly to organic compounds. The distribution into the various kinds of compounds shall be justified.

514. The radioactive substances entering the airspace shall first be assumed to be transported into the environment via the ventilation and filtering system in a way corresponding to the normal functioning of the system. If the ventilation

system can be used in several different ways in the above-mentioned situation, the way leading to the most extensive releases shall be chosen for the analysis.

515. Isolation of ventilation may be assumed in accordance with the design of the plant's protection systems, so that any changes in the parameters used as protection limits during accidents are assessed conservatively.

516. If the pressure and temperature inside the containment during an accident exceed the values for which the containment leak-tightness requirements have been set and during which the leak rate is experimentally measured, the leak rate used for release calculations shall be separately justified."

- Chapter 4.4 'Analyses of the dispersion of releases in licence applications' of Guide YVL C.4 [32] :

"4.4.2 Operational occurrences and accidents

425. In the analyses to justify the technical solutions at the nuclear power plant in the preliminary and final safety analysis report, radioactive releases during operational occurrences and accidents shall be estimated in accordance with Guide YVL B.3.

426. In the assessment of the dispersion of releases caused by operational occurrences and accidents at a nuclear power plant, the effective release height shall be separately selected for each release situation.

427. When selecting dispersion situations for detailed analysis, statistical computation methods shall be used to cover diverse dispersion situations; assessment quantities most significant for each situation shall be evaluated at various distances. The situations and dispersion distances to be analysed in detail shall be chosen such that they represent every operational occurrence, accident, and release height examined. The situations analysed shall include at least those representatives of the average statistical distribution of the quantity (such as dilution factor or total dose) on which the selection was based as well as the upper fractile of 95%. The form of representation and the scope of analysis of the results shall be illustrative and sufficiently versatile."

3. Do you have requirements for identification and categorization of postulated initiating events (covering AOOs, DBAs and DEC/BDBAs)?

According to the definition on the Nuclear Energy Decree (161/1988) [33] and Guide YVL B.1 [34] the categorization is based on frequency as follows:

DiD Level	Category	Frequency per year	Limit
Level 1	Normal operation (DBC 1)		0.1 mSv (annual dose limit for individual of the population)
Level 2	Anticipated operational occurrences (DBC 2)	$f > 10^{-2}$	0.1 mSv (annual dose limit for individual of the population)

Level 3a	Postulated accidents Class 1 (DBC 3)	$10^{-2} > f > 10^{-3}$	1 mSv (annual dose limit for individual of the population)
	Postulated accidents Class 2 (DBC 4)	$f < 10^{-3}$	5 mSv (annual dose limit for individual of the population)
Level 3b	Design extension conditions (Finnish DEC ²) Multiple failures DEC A - CCF combined with DBC2 / DBC3 DEC B - Complex failure combination DEC C - Very rare external event		20mSv (annual dose limit for individual of the population)
Level 4	Severe accidents (SA)	Safety goals: Core Damage Frequency: CDF $< 10^{-5}$ Large Release Frequency: LRF $< 5 \times 10^{-7}$	• The limit for the release of radioactive materials arising from a severe accident is a release which causes neither acute harmful health effects to the population in the vicinity of the nuclear power plant, nor any long-term restrictions on the use of extensive areas of land and water • The requirement applied to long-term effects will be satisfied if there is only an extremely small possibility that, as the result of a severe accident, atmospheric release of cesium-137 will exceed the limit of 100 terabecquerel (TBq)

Nuclear Energy Decree (161/1988) [33] :

“17) anticipated operational occurrence means a deviation from normal operation that can be expected to occur one or more times over a span of one hundred operating years;

18) accident means postulated accidents, design extension conditions and severe accidents;

19) postulated accident means a deviation from normal operation that can be assumed to occur less frequently than once over a span of one hundred operating years, excluding design extension conditions, and which the nuclear facility is required to withstand without sustaining severe fuel failure, even if individual components of systems important to safety are rendered out of operation due to servicing or faults; postulated accidents are grouped into two classes on the basis of the frequency of their initiating events:

a) class 1 postulated accidents, which can be assumed to occur less frequently than once over a span of one hundred operating years, but at least once over a span of one thousand operating years;

b) class 2 postulated accidents, which can be assumed to occur less frequently than once over a span of one thousand operating years;

² According to the Nuclear Energy Decree 161/1988 [33] , design extension conditions is defined as:

“a. an accident where an anticipated operational occurrence or class 1 postulated accident involves a common cause failure in a system required to execute a safety function;

b. an accident caused by a combination of failures identified as significant on the basis of a probabilistic risk assessment; or

c. an accident caused by a rare external event and which the facility is required to withstand without severe fuel failure.”

20) design extension condition means:

- a) an accident where an anticipated operational occurrence or class 1 postulated accident involves a common cause failure in a system required to execute a safety function;
- b) an accident caused by a combination of failures identified as significant on the basis of a probabilistic risk assessment; or
- c) an accident caused by a rare external event and which the facility is required to withstand without severe fuel failure;

21) severe accident means an accident in which a considerable part of the fuel in a reactor or of the spent fuel in a fuel pool or fuel storage loses its original structure”.

Guide YVL B.1 ‘Safety Design of a Nuclear Power Plant’ [34] :

“421a. The third level of defence shall be divided into level 3a and level 3b. At level 3a, the objective is to control the postulated accidents (Class 1 and Class 2) arising from single initiating events and their consequential effects, in order to limit the releases of radioactive materials. At level 3b, the objective is to control design extension conditions so that severe fuel damage can be prevented.

421b. An anticipated operational occurrence or class 1 postulated accident involving a common cause failure in a system required to execute a safety function in safety class 2 shall be regarded as a design extension condition DEC A.

421c. A common cause failure of any individual component type (for example, a similar check valve, same type and manufacturer) shall not prevent the nuclear power plant from being brought to a controlled state or a safe state.

421d. Failure combinations recognised as significant based on probabilistic risk assessment shall be processed as a design extension condition DEC B. Furthermore, such failure combinations or additional failures that emerge during the initiating event, and could significantly affect the integrity of fuel, the radiological effects of the accident or primary circuit pressure shall be considered.

...

423. Events that result in a release requiring measures to protect the population in the early stages of the accident shall be practically eliminated.

423a. Events leading to a large release shall be practically eliminated.

424. Events to be practically eliminated shall be identified and analysed using methods based on deterministic analyses complemented by probabilistic risk assessments and expert assessments. Practical elimination cannot be based solely on compliance with a cut-off probabilistic value. Even if the probabilistic analysis suggests that the probability of an event is extremely low, all practicable measures shall be taken to reduce the risk.”

4. **Would your existing requirements for identification and categorization of PIEs - postulated initiating events (covering AOOs, DBAs and DEC/BDBAs) be applicable to reviews of SMRs including Gen-IV SMRs? If not, why and how would you go about it? Please describe, in your professional opinion (does not have to represent the organizational 'consensus'), what thinking/method would you use to assess PIEs?**

Assessing, identification and categorization of initial events is licensees' responsibility and is made on a plant-specific basis. Neither a specific method nor means for identification and categorization is given in regulatory framework. However, there are requirements regarding assessment of PIEs that could be directly applied to SMR and Gen-IV SMRs under the current legislation. These requirements will be renewed as part of an ongoing regulatory renewal process.

The Guide YVL B.3 [31], in its Chapter 3, defines the events to be analysed as follows:

“301. Analyses pertaining to the plant's behavior as well as releases of radioactive substances and radiation doses caused by the releases shall cover the nuclear power plant's normal operational states, anticipated operational occurrences, postulated accidents, design extension conditions and severe reactor accidents. Examples of the events to be analysed are given in given in GSR Part 4 (Rev. 1) [8] and SSG-2 (Rev. 1)[9].

302. The scope of the analysed events shall provide a comprehensive assessment of the nuclear power plant's behaviour during incidents and accidents as well as releases and doses due to incidents and accidents.

303. Operator actions shall be assessed to identify essential operator actions needed in accident management and to assess the effects of potential operator errors.

304. The inadvertent actuation of every system accomplishing a safety function shall be addressed as an initiating event.

305. Pressure control analyses for the reactor coolant circuit shall consider cases during which the reactor pressure tends to increase or decrease in consequence of an initiating event, and situations where the coolant circuit pressure must be increased or decreased.

306. Pressure control analyses shall cover also low operating temperatures.

308. Severe reactor accident analyses shall cover all actions required for the plant's severe reactor accident strategy and the phenomena associated with the strategy.”

5. **Does your country use a source term only covering specific radionuclides (source term 'list')? If yes, is the list adequate and relevant to SMRs including Gen-IV SMRs?**

No specific source term radionuclides list is defined. All relevant radionuclides must be assessed regarding the plant-specific source term. Categorization/grouping of radionuclides is commonly used in such analyses.

6. **In your country, how are the quantitative safety goals on releases (typically I-131 and Cs-137 are the radionuclides used in quantitative safety goals) defined? Would these quantitative safety goals be applicable to SMRs including Gen-IV SMRs? If not, what would be the regulatory process for an exemption?**

There is a release limit for severe reactor accidents of 100 TBq of Cs-137. This limit aims to ensure that radioactive releases arising from severe reactor accidents do not necessitate large scale protective measures for the public nor any long-term restrictions on the use of extensive areas of land and water. This limit is applied for large scale power plants with certain size emergency planning zone (EPZ) and precautionary action zone (PAZ). In the ongoing renewal of Finnish regulatory framework, this 100TBq CS-137 limit will be reassessed while the regulation concerning the size of PAZ and EPZ has already been changed to be plant-specific. The quantitative safety goals of releases for small and micro scale reactors will be different, possibly through a release limit or based on activity or inventory of the plant. This will be specified during the ongoing regulatory framework renewal process.

APPENDIX B.3 - FRANCE

Nuclear Safety and Radiation Protection Authority (ASNR) Response to Questionnaire on Mechanistic Source Term

1. Do you have regulatory requirements pertaining to mechanistic source terms?

Article 3.7 of the order of 7th February 2012.

I. - The demonstration of nuclear safety includes an assessment of the potential consequences, radiological or not, of the envisaged incidents and accidents. For each scenario, this assessment includes:

- a presentation of the assumptions used in the release calculations and exposure scenarios; the release calculation assumptions must be reasonably pessimistic, and the exposure scenarios must be based on realistic parameters, but without considering any population protection actions that could be implemented by the public authorities;
- an estimation of the effective doses and the intensity of the non-radiological phenomena to which persons and the environment could be exposed in the short, medium and long term, distinguishing the different age classes where necessary, and considering the different hazardous substance transfer pathways; the estimation includes equivalent doses to the thyroid in the event of radioactive substance releases that justify this;
- an estimation of the extent of the areas likely to be affected;
- for incidents or accidents having consequences outside the site, the kinetics of the development of the hazardous phenomena and the propagation of their effects.

...

III. - The intensity of the hazardous radiological phenomena is defined with respect to reference values expressed as levels of intervention of the public authorities in radiological emergency situations, as defined by the ASN in application of article R. 1333-80 of the Public Health Code.

Article 4.7.1 of ASN Resolution 2015-DC-0532 of 17th November 2015 relative to the safety analysis report for basic nuclear installations

For the application of article 3.7 of the abovementioned order of 7th February 2012, the safety analysis report sets out the potential consequences, radiological or not, of the incidents and accidents, with the exception of the accidents mentioned in article 4.4.9 of this appendix. The safety analysis report specifies the following points in particular:

- the hypotheses, rules and methods used to make the evaluation,
- in the case of radiological releases, a description of the physical-chemical form of the released radionuclides contributing the most to the consequences of the releases,

- for incidents or accidents resulting in hazardous or radioactive substances being released into the environment, the direct consequences of the release phase, chiefly associated with atmospheric releases and possible direct releases in the ground and the aquatic environment ,
- for the most representative accidents leading to radioactive releases, the evaluation as a function of time:
 - of the consequences associated with the releases in terms of surface activity and possibly ambient dose rate,
 - of the mass contamination of agricultural commodities and, if applicable, the contamination of water resources.

2. If not, what existing regulatory requirements could be used in your reviews of mechanistic source terms?

Not applicable.

3. Do you have requirements for identification and categorization of postulated initiating events (covering AOOs, DBAs and DEC/BDBAs)?

Article 3.3.1.1.1 of ASN guide n°22 on Design of pressurised water reactors

The determination of the design-basis condition categories (DBC) shall result from:

- the identification of the SIEs whose consequences affect at least one safety function;
- the exclusion of the SIEs for which prevention measures are sufficient (see 3.2.2);
- The grouping of the non-excluded SIEs to define a limited number of design-basis conditions for which the consequences encompass those of the events of the corresponding group.

Article 3.3.1.1.2 of ASN guide n°22 on Design of pressurised water reactors

The SIEs addressed in the demonstration of nuclear safety shall essentially be defined on a deterministic basis taking into account in particular the specificities of the design of the installation in question, the different operating states of the installation and the experience with reactors, whether similar or not, in France or abroad, in operation or under construction.

Article 3.4.1.1 of ASN guide n°22 on Design of pressurised water reactors

The list of events to consider in the design extension envelope shall be based on deterministic and probabilistic considerations, consolidated by expert judgment if necessary.

The events in the design extension envelope shall consider:

- conditions termed "DEC-A" for which fuel meltdown shall be prevented. They consider, in principle, with respect to the fuel meltdown frequency objective:

- combinations of a DBC condition and a common cause failure affecting the redundant parts of an system IP necessary for the control of this DBC condition;
- common cause failures on the systems IP used in normal operation.

The probabilistic analyses enable the list of DEC-A conditions (see article 6.3.1) to be confirmed and supplemented if necessary;

- conditions termed "DEC-B" in which fuel meltdown is postulated despite the measures taken to prevent this.

4. Would your existing requirements for identification and categorization of PIEs - postulated initiating events (covering AOOs, DBAs and DEC/BDBAs) be applicable to reviews of SMRs including Gen-IV SMRs? If not, why and how would you go about it? Please describe, in your professional opinion (does not have to represent the organizational 'consensus'), what thinking/method would you use to assess PIEs?

Yes. These are generic requirements.

5. Does your country use a source term only covering specific radionuclides (source term 'list')? If yes, is the list adequate and relevant to SMRs including Gen-IV SMRs?

No. The licensee should consider a large number of isotopes. It could depend on the reactor technology.

6. In your country, how are the quantitative safety goals on releases (typically I-131 and Cs-137 are the radionuclides used in quantitative safety goals) defined? Would these quantitative safety goals be applicable to SMRs including Gen-IV SMRs? If not, what would be the regulatory process for an exemption?

Article 3.3.1.5.3 of ASN guide n°22 on Design of pressurised water reactors

The assessment of the radiological consequences of events in the design reference envelope contributes to the verification of the adequacy of the design provisions adopted with respect to the safety objectives mentioned in chapter II.1 of this guide. Criteria for assessing the consequences for humans and the environment and enabling these objectives to be satisfied shall be defined for each design-basis conditions category.

Example of dose limits for large NPP.

Category		Dose level
Normal operation	DBC-1	- Workers : 20mSv/y - Populations : 1mSv/y
Incidental condition	DBC-2	
Accidental condition	DBC-3	Nearest off-site population (or generic distance of 500 m): - Effective dose : 10 mSv/7d ; - Equivalent dose to the thyroid : 50 mSv/7d → No off-site countermeasures
	DBC-4	
Design extension conditions	DEC-A	
	DEC-B	

APPENDIX B.4 - JAPAN

Nuclear Regulation Authority (NRA) Response to the Questionnaire on Mechanistic Source Term³

1. Do you have regulatory requirements pertaining to mechanistic source terms?

For commercial LWRs

No. The NRA does not have requirements for the establishment of the source term based on analysis of the amount of damage to the cladding, fuel, and core.

For test and research reactors (including HTGR)

No. The regulatory standards for test and research reactors do not include regulatory requirements for MSTs.

2. If not, what existing regulatory requirements could be used in your reviews of mechanistic source terms?

For commercial LWRs

In the event of a severe accident, the total amount of radioactive materials released is required to have as little impact on the environment as possible, and in consideration of the impact on residents living near the nuclear power plant, the amount of Cs-137, which has a long half-life with a large assumed release volume, is confirmed to be less than 100 TBq⁴. In addition, in the regulatory review it is confirmed if the dose assessments for radiation exposure at emergency response centers, etc. - used as a basis for response to accidents - are conducted using source terms comparable to those from the Fukushima Daiichi NPS accident.

The relevant review criteria and guidelines are as follows.

- Article 37 'Prevention of severe accidents and their escalation' of the 'NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Commercial Power Reactors and their Auxiliary Facilities' [35] :

"2-2 Power reactor facility shall be such as to be provided with measures necessary to prevent failures of containment vessel and abnormal release of radioactive materials to the outside of the station" prescribed in para. (2) means that the following requirements shall be met.

(a) It shall be verified that the measures to prevent failures of the containment vessel and release of radioactive materials at an abnormal level to the outside of the site are effective against the postulated containment vessel failure modes.

³ Please note the text of the regulatory standards included in the responses to the questionnaire is a provisional English translation.

⁴ Since the total amount of Cs-137 released to the environment in the Fukushima Daiichi NPP accident is evaluated to have been about 10,000 TBq based on the analysis results, it is based on the idea that if the total amount of Cs-137 released is about 100 TBq, the impact of contamination by radioactive materials on the environment can be reduced.

2-3 "The measures are effective" described in 2-2 means that the following evaluation items are met for the most part.

...

(c) The total release of radioactive materials shall be controlled to minimize the impact on the environment in consideration of environmental contamination by radioactive materials."

- Article 42 'Specified severe accidents response facilities' of the 'NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Commercial Power Reactors and their Auxiliary Facilities [35]

"(d) In the event that measures to prevent failures of a containment vessel taken by SA response facilities (excluding those that constitute the specified SA response facilities) do not function effectively, the following habitability requirements of the emergency control room shall be satisfied, assuming that the operations performed in the reactor control room have moved to the emergency control room:
(i) The postulated released amount of radioactive materials shall be equivalent to that at the time of the accident at the Fukushima Daiichi Nuclear Power Plant;"

- Article 61 'Emergency response center' of the 'NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Commercial Power Reactors and their Auxiliary Facilities' [35]

"e) The following requirements need to be satisfied for the habitability of the emergency response center.

(1) The postulated amount of released radioactive materials and other figures shall be the same as those in the accident at the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company."

- Review Guide for Evaluation of the Effectiveness of Core Damage Prevention Measures and Containment Damage Prevention Measures for Commercial Power Reactors [36] :

"(6) In order to confirm that "the total release of radioactive materials shall be controlled to minimize the impact on the environment in consideration of environmental contamination by radioactive materials", as stipulated in Article 37, 2-3(c) of Regulatory Guide of the NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Commercial Power Reactors and their Auxiliary Facilities, for the assumed containment failure mode, the amount of Cs-137 released shall be 100 TBq for the assumed containment failure mode."

- Review Guide for Radiation Exposure Assessment of Habitability of Control Rooms and Emergency Response Centers in the Event of a Severe Accident for Commercial Power Generation Reactors⁵

⁵ <https://www.nra.go.jp/data/000069142.pdf>

"4 Standard Methodology for Assessing Exposure related to Habitability

4.1 Methodology and Scope of Exposure Assessment for Habitability

...

(2) Evaluation procedure

...

a. Establish a source term to be used for the evaluation of exposure for habitability of the reactor control room/emergency control room/emergency response center

- In the exposure assessment for the habitability of the reactor control room, the amount of radioactive material released to the atmosphere and the distribution of radioactive material presence in the reactor facility are set based on the source term analysis of the accident sequence with the most severe results in terms of exposure of the reactor control room operators or countermeasure personnel among the accident sequences with successful accident convergence in the containment failure mode assumed in the effectiveness assessment of containment failure prevention measures (In this case, the containment is sound because the containment failure prevention measures are effective).
- In the exposure assessment for the habitability of the emergency control room or emergency response center, the amount of radioactive materials discharged into the atmosphere is calculated from the release rate of radioactive materials into the atmosphere and the amount of radioactive materials in the reactor core for an accident assuming that the release rate of radioactive materials into the atmosphere is equivalent to the accident at the TEPCO's Fukushima Daiichi NPP. The distribution of the amount of radioactive materials present in the reactor facility is set based on the rate of release of radioactive materials into the reactor containment vessel and the amount in the reactor core.

...

4.4 Main analysis conditions, etc. for exposure assessment related to habitability of emergency control room or emergency response center

(1) Source term

a. Rate of release to the atmosphere

The ratio of radioactive materials released into the atmosphere to the amount in the core immediately before the accident is assumed to be the same as that of the accident at the TEPCO's Fukushima Daiichi NPS, where the reactor containment vessel is considered to have been breached.

Noble gases: 97%

Iodine: 2.78%

(CsI: 95%, inorganic iodine: 4.85%, organic iodine: 0.15%)

(Set referring to NUREG-1465)

Cs: 2.13%

Te: 1.47%

Ba: 0.0264%

Ru: $7.53 \times 10^{-8}\%$

Ce: $1.51 \times 10^{-4}\%$

La: $3.87 \times 10^{-5}\%$ (%)

b. Start time and duration of release of radioactive materials

The start time of the release of radioactive materials into the atmosphere is assumed to be 24 hours after the occurrence of the accident (reactor scram) (the

start time of the release of radioactive materials was set with reference to that of the No.1 reactor that first released radioactive materials in the accident at TEPCO's Fukushima Daiichi Nuclear Power Station).

The duration of radioactive material release into the atmosphere is assumed to be 10 hours in order to obtain conservative results (set with reference to the duration of the release from the No. 2 reactor at the TEPCO's Fukushima Daiichi NPS)."

Example regulatory review

- Review Results for Application for Permission for Changes to Nuclear Reactor Installation for Reactor No. 2 of Shimane Nuclear Power Station⁶

"In order to confirm the effectiveness of containment failure prevention measures in this containment failure mode, the applicant has selected the evaluated accident sequence, the analysis code, and the analysis conditions as follows.

...

Cs-137 leaks from the reactor containment vessel and is released to the environment via the reactor building. The amount of leakage from the reactor pressure vessel into the reactor containment vessel is assumed to be based on the rate at which Cs-137 contained in the core leaks into the reactor containment vessel according to the progress of the event. It is calculated using MAAP, which tends to show a larger release rate than that of NUREG-1465, a report on a typical source term. For the removal effect of Cs-137 in the reactor containment vessel, the effect of the reactor containment vessel spray and suppression chamber scrubbing with pool water, etc., is considered. The leakage rate from the reactor containment vessel to the reactor building is evaluated based on the design leakage rate and the pressure in the reactor containment vessel. The decontamination factor due to the collection effect of particulate radioactive materials at the reactor containment vessel penetration, etc. is assumed to be 10.

...

The effective dose of radiation exposure of the operators and emergency response personnel was evaluated assuming that the assumed emitted amount of radioactive materials was the same as that of the accident at the TEPCO's Fukushima Daiichi NPS, and without considering conditions such as the wearing of masks, personnel rotation system, taking of stabilized iodine tablets, temporary facilities, etc. The result was confirmed as approximately 1.7 mSv in 7 days."

For test and research reactors

The High Temperature Engineering Test Reactor (HTTR), the first HTGR constructed in Japan at the Oarai Research and Development Institute of the Japan Atomic Energy Agency (JAEA) with 30 MW in thermal power and 950°C of outlet coolant temperature, is regulated as a test and research reactor.

For evaluating the exposure effects of a design-basis accident, the regulatory standard NRA Ordinance Prescribing Standards for the Location, Structure, and

⁶ <https://www.nra.go.jp/data/000365227.pdf>

Equipment of Reactors Used for Testing and Research [37] , on its article 13, item (ii) subitem (c), requires that the testing and research reactor facilities “shall not cause radiation hazards to the public in the vicinity of the plant, etc.” and that the specific details of the evaluation of the radioactive material release shall be in accordance with the Review Guidelines for Safety Evaluation of Water-cooled Test and Research Reactor Facilities [38] . However, for water-cooled research reactors with thermal power exceeding 50 MW, the ‘Review Guideline regarding Safety Evaluation of Light Water Reactor Facilities for Power Generation’⁷ is to be referred to. For reactors other than water-cooled research reactors, such as gas-cooled reactors and sodium-cooled fast reactors, Ref. [38] may be used as a reference, except for matters specific to water-cooled reactors.

For example, the review guide [38] provides the following regarding the evaluation of radioactive material release.

- In the calculation of public doses, the contribution of radionuclides other than iodine and noble gases to atmospheric emissions is considered to be negligible; therefore, in principle, the focus should be on iodine and noble gases.
- The effect of removal of radioactive materials in water should be evaluated and set appropriately. For the transport process of radioactive materials, the influence of reactor coolant leakage should be appropriately evaluated.
- When expecting the emergency exhaust system to operate, the case of loss of external power should be considered.

For accidents that occur less frequently than the design basis accidents but could cause excessive radiation exposure to the public in the vicinity of the site (an evaluated effective dose exceeding 5 millisieverts per accident), evaluation and countermeasures are required, and specific evaluation is confirmed during the regulatory review (Articles 40 and 53 of Ref. [37]).

Example of regulatory review practices for HTTR

Application for Modification of HTTR Nuclear Facility [39]

1) Design Basis Accident⁸

The following is an example of conditions for evaluating radioactive material release by MST for ‘the primary cooling system double pipe break accident’, which is one of the design basis accidents reviewed for HTTR.

- It is assumed that 1% of all coated fuel particles in the core have undergone coating failure before the design basis accident, and that the coated fuel particles with coating failure are uniformly distributed in the core.
- The radioactive materials released into the containment vessel are assumed to be all the fission products circulating in the primary cooling system and 60% of the radioactive iodine deposited on the walls of the primary cooling system, which

⁷ <https://warp.da.ndl.go.jp/info:ndljp/pid/9483636/www.nsr.go.jp/archive/nsc/shinsashishin/pdf/1/si008.pdf>

⁸ para. 10-3-17 of [39]

are released into the containment vessel instantaneously after a design basis accident.

- Of the iodine released into the containment vessel, organic iodine accounts for 4%, and the remaining 96% is inorganic iodine.
- Of the radioactive iodine released into the containment vessel, 50% of the inorganic iodine adheres to the containment vessel and equipment within the containment vessel and does not contribute to leakage from the containment vessel. Organic iodine and noble gases are not expected to have this effect.
- The leakage rate from the containment vessel is 0.25%/d, which is higher than the design leakage rate of 0.1%/d, taking into account the pressure and temperature in the containment vessel in the event of a design basis accident.
- Fission products leaked from the containment vessel into the reactor building are released through the exhaust pipe via the emergency air purification system. It is assumed that there is no effect of deposition or adhesion of fission products in the reactor building.
- The fission products discharged from the reactor building to the ground are set in consideration of the pressure difference between the reactor building and atmospheric pressure, the design flow rate of the emergency air purification system, and its operation time.
- The iodine removal efficiency of the filter of the emergency air purification system is set to 90% as a value with a margin of the design value (95%).

2) Accident that releases large amounts of radioactive materials⁹

The following is an example of conditions for evaluating radioactive material release by MST for 'primary cooling system double pipe break accident', which is one of the accidents that release large amounts of radioactive materials of the HTTR.

- The fuel failure rate is assumed to be the same as in the design basis accident.
- The loss of containment function of the reactor containment vessel is assumed to be caused by the superimposition of an earthquake equivalent to the basic earthquake ground motion and the loss of function of the reactor containment vessel, emergency air purification system, and emergency generator, which are designed for Class B seismic resistance.
- A reactor containment failure is defined as the loss of the containment confinement function due to an opening equivalent to the inner diameter of the penetrating piping of the containment vessel caused by a rupture of the entire circumference of the piping, resulting in an instantaneous release of radioactive materials from the containment vessel into the reactor building.
- Consideration is given to the use of heat-resistant tape to seal cracks in the reactor building in order to restore airtightness, avoid the release of fission products on the ground, and promote the release of radioactive materials from high places in the building, thereby reducing the concentration of radioactive materials in the vicinity of the site.
- The time decay of radioactive materials is taken into account.

⁹ para. 10-4-5 of [39] and <https://www2.nra.go.jp/data/000470082.pdf> (p. Article 53-188)

3. Do you have requirements for identification and categorization of postulated initiating events (covering AOOs, DBAs and DEC/BDBAs)?

Among the postulated initiating events for test and research reactors, AOOs and DBAs are presented in the 'Review Guidelines regarding Safety Evaluation of Water-cooled Test and Research Reactor Facilities'[38] , which is cited in the regulatory standards.

In the Guidelines, the following events are listed as specific events for AOOs and DBAs.

I. AOOs

- Abnormal changes in reactivity or power distribution in the core
 - Abnormal control rod withdrawal during reactor startup
 - Abnormal control rod withdrawal during power operation
 - Addition of reactivity due to injection of cold water (reactors with plate fuel)
 - Addition of reactivity due to abnormalities in experimental materials, etc.
- Abnormal change in heat generation or heat removal in the core
 - Decrease in flow rate of primary coolant (reactors with thermal output of 500 kW or more)
 - Decrease in secondary coolant flow (reactors with thermal output of 500 kW or more)
 - Loss of commercial power supply
- Other events deemed necessary by the reactor facility design
 - a. Abnormal change in pressure in the primary coolant system (pressurized reactors)
 - b. Light water inflow into the heavy water reflector (reactors with heavy water reflector)
 - c. Abnormalities in temperature, pressure, etc. of the experimental facility and reactor due to malfunction of the experimental facility (reactors with in-core loop experimental facility, etc.)
 - d. Other abnormalities deemed necessary by the design of the reactor facility

II. DBAs

- Abnormal reactivity input
 - Fuel drop or fuel misloading
 - Abnormal addition of reactivity due to damage of experimental facilities, etc. (reactors with in-pile loop experimental facilities, etc.)
- Reactor coolant leakage or significant change in core cooling conditions
 - Reactor coolant leakage
 - Failure of the reactor coolant pump shaft (reactors with thermal output of 500 kW or more)
 - Blockage of the reactor core flow path (reactors using plate fuel)

- Abnormal release of radioactive materials into the environment
 - Mechanical failure of spent fuel
 - Significant damage of experimental facilities, experimental objects, etc. (Reactors with experimental facilities)
 - Damage to fuel, etc. due to abnormal input of reactivity, leakage of reactor coolant, or significant change in core cooling conditions
- Other events deemed necessary according to the design of the reactor facility
 - Damage to radioactive waste treatment facilities
 - Other abnormalities deemed necessary by the design of the reactor facility

Example of regulatory review practices for HTTR

Then, for the HTTR, the factors affecting each item of the judgment criteria were extracted and analyzed, and abnormal events were identified, referring not only to the said examination guidelines but also to the 'Regulatory Guide for Evaluating Safety Assessment of Light Water Reactor Facilities'[40] In this process, the events were classified and evaluated according to their nature, and the event with the most severe evaluation result in terms of the judgment criteria was selected as the representative event¹⁰.

I. AOOs¹¹

- Abnormal changes in reactivity or power distribution in the core
 - Control rod withdrawal from subcritical state
 - Control rod withdrawal during power operation
- Abnormal change in heat generation or heat removal in the core (including abnormal changes in reactor coolant pressure or amount of reactor coolant retained)
 - Decrease in primary coolant flow rate
 - Increase in primary coolant flow rate
 - Decrease in heat removal in the secondary cooling system
 - Increase in heat removal from secondary cooling system
 - Loss of commercial power supply
- Anomalies in irradiated samples and experimental facilities
- Abnormality during special operation

II. DBAs¹²

- Loss of core cooling capacity
 - Accidents involving blockage of coolant flow paths in the fuel body
 - Primary cooling system tube failure accident
 - Secondary helium cooling system tube failure

¹⁰ paras. 10-1-1, 10-3-39 of [39]

¹¹ para. 10-2-1 of [39]

¹² para. 10-3-1 of [39]

- Secondary helium cooling system double-pipe rupture accident
- Pressurized water pipe rupture accident
- Depressurization accidents
 - Primary cooling system double pipe rupture accident
- Water intrusion accidents
 - Primary pressurized water cooler heat-transfer tube breakage accident
- Accidents involving primary helium purification equipment
 - Primary helium purification facility breakage accident
- Accidents related to waste treatment facilities
 - Gas waste disposal facility damage accident
- Accidents related to irradiation samples and experimental facilities
 - Accident in which sweep gas piping of irradiation test equipment was damaged
 - Accident involving blockage of coolant flow path for fuel-limited irradiation specimens
- Accidents due to control rod breakage
 - Standpipe breakage accident

III. DECs/BDBAs

In addition, for DECs/BDBAs of gas-cooled reactors among test and research reactors, the Article 53 of the Regulatory Guide [37] specifies the evaluation and countermeasures for accidents that may cause transient radiation exposure to the public in the vicinity of the site (with an evaluated effective dose exceeding 5 mSv per occurrence accident), although the frequency of occurrence is lower than design basis accidents. The following example is presented in the Guide for specific accidents for which evaluation and countermeasures are required:

- Oxidation of the in-vessel structure (graphite) and explosion of combustible gases due to air or water intrusion exceeding the design basis accident assumptions.

Considering the characteristics of the HTTR facility that no large amount of radioactive materials is released and there is no oxidation of graphite due to air ingress or explosion due to flammable gases unless the reactor coolant pressure boundary is damaged, the applicant has chosen DECs/BDBAs assuming the superimposition of 'primary cooling system double pipe rupture accident', which causes the largest fracture size in the reactor coolant pressure boundary among DBAs, and the loss of each safety function, shutdown, cooling, and confinement¹³.

4. **Would your existing requirements for identification and categorization of PIEs - postulated initiating events (covering AOOs, DBAs and DECs/BDBAs) be applicable to reviews of SMRs including Gen-IV SMRs? If not, why and how would you go about**

¹³ para. 10-4-1 of [39]

it? Please describe, in your professional opinion (does not have to represent the organizational 'consensus'), what thinking/method would you use to assess PIEs?

Although the current approach for determining and classifying postulated initiating events may be useful for some types of SMRs, it needs to be re-examined considering the characteristics of the facility and its safety design of a new reactor type.

In the review of Japanese nuclear reactors for testing and research, NRA have requested the applicant to systematically analyze the selection of AOOs and DBAs using Failure Mode and Effects Analysis (FMEA), which is widely used to identify potential failures in general industrial fields, to confirm that the selection of AOOs and DBAs has been properly performed without any omissions.

In addition, regarding event selection for DEC/BDBAs, the NRA has requested that the applicant utilize probabilistic risk assessment (PRA) based on the concept of defense in depth.

These methods may be applicable to SMRs if the conditions are met.

5. Does your country use a source term only covering specific radionuclides (source term 'list')? If yes, is the list adequate and relevant to SMRs including Gen-IV SMRs?

For commercial LWRs

- Section 2 'Evaluation Guide for the Effectiveness of Core Damage Prevention Measures' of the Ref. [36] states that "the effectiveness evaluation of accident sequence groups using a containment pressure relief device shall evaluate the effective dose at the site boundary and confirm that there is no significant risk of radiation exposure to the public near the accident site (approximately 5 mSv or less per accident). However, no specific 'list' of radioactive materials is specified.
- In Section 3 'Evaluation Guide for the Effectiveness of Containment Damage Prevention Measures' of the guide, it is stated that "to confirm that the environmental impact is kept as small as possible, including from the viewpoint of contamination of the environment by radioactive materials", as stipulated in Article 37-2-3(c) of the 'NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Commercial Power Reactors and their Auxiliary Facilities' [35], it should be confirmed that the amount of Cs-137 released is less than 100 TBq.

For test and research reactors

- For test and research reactors, fission products are evaluated on a case-by-case basis based on the type of fuel used (uranium fuel, plutonium/uranium mixed oxide), the enrichment of uranium or plutonium, and the burnup of the fuel. They are also evaluated based on the characteristics of the facility. For example, for sodium-cooled fast reactors, the evaluation also takes into account the fact that sodium, as a coolant, has a certain degree of effectiveness in reducing the release of Cs into the environment.

6. In your country, how are the quantitative safety goals on releases (typically I-131 and Cs-137 are the radionuclides used in quantitative safety goals) defined? Would these quantitative safety goals be applicable to SMRs including Gen-IV SMRs? If not, what would be the regulatory process for an exemption?

Regarding the discussion of safety goals at the NRAJ, "safety goals are goals that the NRAJ aims to achieve in the course of its regulation of nuclear facilities. The results of the detailed discussions that had been conducted by the Nuclear Safety Commission's Special Committee on Safety Goals by 2006, such as "about 10^{-4} /year" for the frequency of core damage and "about 10^{-5} /year" for the frequency of loss of containment function, are considered to be sufficient basis for the Nuclear Regulation Authority in discussing safety goals. However, in light of the accident at TEPCO's Fukushima Daiichi NPS, it is necessary to incorporate the perspective of contamination of the environment by radioactive materials into the safety goals, and to keep the environmental impact as small as possible in the event of an accident. Specifically, with reference to examples from other countries around the world, it should be added that for power reactors, the frequency of accidents in which the release of cesium-137 exceeds 100 terabecquerels in the event of an accident should be controlled to not exceed about once in a million reactor years, excluding accidents caused by terrorism and the like. ... And, in light of the purpose of introducing backfit regulations under the Nuclear Reactor Regulation Law, the safety goals should be applied to all power reactors without distinction under the current circumstances. The Nuclear Regulation Authority, which aims to continuously improve safety, will continue to discuss safety goals, including discussions on remaining issues (e.g. whether target values should be separated for new reactors and existing reactors) among those submitted to the Nuclear Regulation Authority on March 6, 2013." The report goes on to state. (see paras. 83, 84 of 'Concept of New Regulatory Standards for Commercial Power Reactors'¹⁴)

¹⁴ <https://www.nra.go.jp/data/000155788.pdf>

APPENDIX B.5 - NETHERLANDS KINGDOM OF

Authority for Nuclear Safety and Radiation Protection (ANVS) response to the
Questionnaire on Mechanistic Source Term

1. Do you have regulatory requirements pertaining to mechanistic source terms?

The Decree on Nuclear Facilities, Fissionable Materials and Ores[41] (Dutch: Besluit Kerninstallaties, Splijtstoffen en Ertsen, Bkse), in Chapter II, § 3, Article 6, lays down more detailed specifications of the information that shall be included in applications for a licence. Para. 1, sub i requires that a risk analysis for beyond design accidents shall be included in the application. Article 18, para. 3, of the Bkse Decree stipulates that a licence for a nuclear facility may be refused if the results from the risk analysis of beyond design basis accidents do not meet the prescribed individual risk (a) and group risk (b) limit values:

- a probability of 10^{-6} per year that a person residing permanently and unprotected outside the relevant facility will die as a result of a beyond design basis accident;
- a probability of 10^{-5} per year that a group of at least ten people present outside the relevant facility will be direct fatalities of a beyond design basis accident, or a n^2 times smaller probability for n times more direct fatalities.

Article 3a (para. 2) of the Bkse Decree stipulates that the government can lay down rules for the manner in which a risk analysis (such as a PSA) shall be performed. For the determination of the necessary environmental dose equivalents and the equivalent and effective doses, the ANVS Regulation on Basic Safety Standards for Radiation Protection (Vbs) applies the *Computation rules for analysis of the consequences of ionizing radiation* (AGIS) for what is referred to as the 'closer examination' laid down in Annex 10 accompanying Article 4.37-4.39 of the ANVS Regulation (Vbs).

The ANVS Regulation (Vbs) refers for the 'closer examination' of releases to the atmosphere and water to the report *Calculation of the dose in the environment on granting an ionizing radiation licence - Part A: releases to the atmosphere and water* (DOVIS-A). That report discusses the calculation of the dispersion and deposition of radioactive materials in the environment and the calculation of the resulting doses. DOVIS-A prescribes the hour-by-hour method of the New National Model (NNM) as the standard for calculation of the dispersion of radioactive material in the atmosphere. The NNM can be used to model the dispersion of air pollution, for example within the context of environmental permits. ANVS has published a Level 3 PSA guideline[42] which is in line with the models used in DOVIS-A and, for releases to the atmosphere, the NNM.

The use of the computation rules for the 'closer examination' of routine emissions is required by law. The use of those computation rules is not prescribed for the calculation of doses and probabilities for testing against the Bkse Decree criteria, in

part because some of those rules are not suitable for application in accident analyses. This received a great deal of attention during the preparations of the present Guide, and where necessary computation rules for level 3 PSA have been formulated that fit in as closely as possible with the original computation rules.

A risk analysis of beyond design basis accidents in which the results shall be tested against the limit values prescribed above is performed on the basis of the level 3 PSA specified in the ANVS Guide on Level 3 PSA (2020). Level 1 and 2 should be based on IAEA standards SSG-3 and SSG-4.

Level 2 PSA determines the probability of the failure of the reactor containment building and the characteristics of the associated accident releases (the source term). Analyses of thermohydraulic and chemical processes play an important role in a level 2 PSA. It results in a 'source term spectrum'. In practice for the existing NPP there are about 20 spectra (including some for the spent fuel pool) representing a certain emission category with a frequency (the total being the CDF), moment of release (early or late) and a release fraction of the nuclides in the core. In fact, each emission category is based on a representing PIE. MAAP is used to generate the source terms.

2. If not, what existing regulatory requirements could be used in your reviews of mechanistic source terms?

See Response to Q1.

3. Do you have requirements for identification and categorization of postulated initiating events (covering AOOs, DBAs and DEC/BDBAs)?

For the design of the NPP the DSR (guidance document Dutch Safety Requirements) is relevant. In the DSR chapter 2 the table 2-1 Technical Safety Concept describes the levels of DiD related to the categories NO, AOO, Single and multiple failure events and core melt accidents. But no frequencies are mentioned.

For the comparison of the emissions to the environment with the dose limits, the legal framework (Bkse Decree) requires that it shall be demonstrated that the following dose limit values prescribed in Article 18, para. 2, of the Bkse Decree are not exceeded:

Event frequency F per year	Maximum effective dose allowed		Maximum thyroid dose allowed
	people 16 years and over	people under the age of 16	
$F \geq 10^{-1}$	0.1 mSv	0.04 mSv	500 mSv
$10^{-1} > F \geq 10^{-2}$	1 mSv	0.4 mSv	500 mSv
$10^{-2} > F \geq 10^{-4}$	10 mSv	4 mSv	500 mSv
$F < 10^{-4}$	100 mSv	40 mSv	500 mSv

The first column shows a frequency categorization of the initiating events, but no direct relation with levels of DiD.

4. **Would your existing requirements for identification and categorization of PIEs - postulated initiating events (covering AOOs, DBAs and DEC/BDBAs) be applicable to reviews of SMRs including Gen-IV SMRs? If not, why and how would you go about it? Please describe, in your professional opinion (does not have to represent the organizational 'consensus'), what thinking/method would you use to assess PIEs?**

The Law and DSR can be applied to SMRs as well. Since the DSR is guidance, well justified deviations, alternatives and a graded approach may be proposed by the applicant: it is comply or explain. It is expected that Gen III/III+ can be handled more easily than Gen IV reactors, where concepts such as core melt or containment may be approached or defined differently.

5. **Does your country use a source term only covering specific radionuclides (source term 'list')? If yes, is the list adequate and relevant to SMRs including Gen-IV SMRs?**

As in question 1 we have a source term spectrum from PSA Level 2. For each of the about 20 spectra a different set of radionuclides is applied that are relevant. For SMR (gen III/III+ and gen IV) the applicant should develop the source term spectra relevant for that installation. It depends on the design if comparable sets could be used (maybe easier for Gen III/III+). For gen IV it is expected to be different, because certainly the PIEs and source term spectra could be different.

6. **In your country, how are the quantitative safety goals on releases (typically I-131 and Cs-137 are the radionuclides used in quantitative safety goals) defined? Would these quantitative safety goals be applicable to SMRs including Gen-IV SMRs? If not, what would be the regulatory process for an exemption?**

There exist no other targets than those in the law. In the DSR is one target: at level 4 of defense in depth, accidents with core melt. The total probability of occurrence of accidents with core melts shall be as low as reasonably achievable but shall not exceed 10^{-6} per year.

APPENDIX B.6 - REPUBLIC OF KOREA

Korea Institute of Nuclear Safety (KINS) Unofficial response to the Questionnaire on Mechanistic Source Term

1. Do you have regulatory requirements pertaining to mechanistic source terms?

No, we do not.

2. If not, what existing regulatory requirements could be used in your reviews of mechanistic source terms?

We have allowed only traditional source terms (RG 1.195 and TID-14844).

3. Do you have requirements for identification and categorization of postulated initiating events (covering AOOs, DBAs and DEC/BDBAs)?

Postulated Initiating events are classified by referring to RG 1.70, ANS technical standard, etc.

4. Would your existing requirements for identification and categorization of PIEs - postulated initiating events (covering AOOs, DBAs and DEC/BDBAs) be applicable to reviews of SMRs including Gen-IV SMRs? If not, why and how would you go about it? Please describe, in your professional opinion (does not have to represent the organizational 'consensus'), what thinking/method would you use to assess PIEs?

Case classifications are generally applicable. However, due to the design characteristics of SMR, some systems/components are often excluded. Therefore, some accidents need to be discussed on a case-by-case basis.

5. Does your country use a source term only covering specific radionuclides (source term 'list')? If yes, is the list adequate and relevant to SMRs including Gen-IV SMRs?

The list of source terms can be found in RG 1.195 Table 3. In the case of RG 1.183, there is a slightly more extended list of source terms. There is no problem with applying it to light water type SMR (SMART100, etc.). In the case of Gen-4 (such as molten salt reactor) has not been considered.

6. In your country, how are the quantitative safety goals on releases (typically I-131 and Cs-137 are the radionuclides used in quantitative safety goals) defined? Would these quantitative safety goals be applicable to SMRs including Gen-IV SMRs? If not, what would be the regulatory process for an exemption?

The notice of NSSC (2017-34) 'Regulations on the scope of accident management and the detailed criteria for evaluating accident management capabilities' stipulates that the total sum of accident frequencies that release the radionuclide Cs-137 over 100 TBq shall not exceed 1.0×10^{-6} /year as a target value to be applied to probabilistic safety assessment. We are currently conducting research on reorganizing the

regulatory framework to reflect the design characteristics of SMRs, so we plan to review and decide whether it is necessary to change the safety goals.

APPENDIX B.7 - RUSSIAN FEDERATION**Scientific and Engineering Centre for Nuclear and Radiation Safety (SEC NRC) Unofficial response to the Questionnaire on Mechanistic Source Term****1. Do you have regulatory requirements pertaining to mechanistic source terms?**

No.

2. If not, what existing regulatory requirements could be used in your reviews of mechanistic source terms?

The requirements of regulatory documents of the Russian Federation (federal rules and regulations in the field of the use of atomic energy, sanitary norms and rules) do not use term mechanical source term or similar. The requirements for determining the source term during a safety analysis are common for NPPs/SMRs in the federal rules and regulations in the field of atomic energy use NP-001-15[43] and NP-006-16[44] .

Para. 1.2.14, 1.2.16 NP-001-16[44] and para. 15 of Appendix No. 3 to NP-006-16[44] establish that the safety analysis report for NPPs/SMRs with VVER-type reactors should contain an analysis of design basis and beyond design basis accidents, considering all possible initiating events not excluded by the principles of the reactor design. This analysis should consider the single failure principle and the possible operability or inoperability of safety systems and special technical means for managing beyond design basis accidents. Also, para. 15 of Appendix No. 3 to NP-006-16[44] establishes that the analysis of design basis and beyond design basis accidents should contain not only an analysis of the events that occurred (failures), but also an assessment of the source term.

Para. 1.8.3 of NP-006-16[44] requires the following information on the PSA results to be provided:

- types of PSA performed (PSA of Level 1, PSA of Level 2);
- information on consideration of all locations of NM, RS and RW at the NPP;
- information on consideration of all initial states of the NPP normal operation (operation at full and partial power level; refueling, warming and cooldown states);
- description of the applied reliability database;
- information on the list of considered initiating events (internal IEs, fires and floods as well as external impacts of natural and human-induced origin);
- information on the quantitative reliability analysis performed for the systems and consideration of interrelations between the systems;
- information on the applied accident sequence models, success criteria for performance of the required functions of the main systems;
- information on consideration of common cause failures;
- information on consideration of personnel actions and errors;

- information on sensitivity and uncertainty assessments;
- final PSA results with information on compliance with the requirements of the federal rules and regulations governing the basic requirements for probabilistic safety assessment of the NPP power unit.

Information on the main contributors to probability of a severe accident and probability of a large-scale emergency release and distribution of their relative contribution shares shall be specified.

Para. 3 of Appendix No. 8 to NP-006-16[44] establishes that a source term assessment for NPPs/SMRs with VVER-type reactors should be presented in the safety analysis report as a function of time for various stages of the accident (i.e. considering stage-specific events and failures).

Para. 1.2.7 of NP-001-15[43] establishes that decisions made to ensure the safety of nuclear power plants should be tested by previous experience, tests, research, and operating experience of prototypes. From this requirement it follows that the parameters used in source term assessing (for example, the proportion of radioactive substances entering the air from a spill or scattering of radioactive substances) should be based on experimental data.

Para. 1.2.9 of NP-001-15[43] establishes that the analysis of design basis accidents (thereinafter - DBA) at NPPs/SMRs (including source term assessment) should be carried out based on a conservative approach.

Also, para. 1.2.9 of NP-001-15[43] establishes that computer programs used in the analysis of accidents at NPPs/SMRs (including source term assessment) should be certified. Certification of a computer program means verification and validation of the program and approval of the results of verification and validation by the regulatory body.

Para. 1.2.16 of NP-001-15[43] also establishes that the analysis of beyond design basis accidents (thereinafter - BDBA) at nuclear power plants (including source term assessment) should be carried out based on a realistic approach. This approach includes determining the most expected value of a parameter important for the safety of nuclear power plants when performing design basis analyses of beyond design basis accidents at nuclear power plants. According to RB-166-20[47] In order to assess the error of the calculation results of the computer program during its validation it is recommended to use both experiments on individual processes and phenomena and integral experiments performed under conditions close to the expected conditions at the plant during a beyond design basis accident. The result of computer program validation is an assessment of the component of calculation error associated with simplifications and approximations of physical and mathematical models implemented in the computer program. The proposed approach is based on the methodology described in the ASME V V 20 - 2009 (R2021) standard. However, the principal feature of the analysis of NPP safety in terms of severe accidents is the absence of measurements at the natural object in the analyzed modes.

The specified requirements of NP-001-15[43] and NP-006-16[44] allow the regulatory body to fully evaluate the source term assessments.

The recommendations of RB-166-20[47] can be used as a basis for the implementation of mechanistic source terms.

3. Do you have requirements for identification and categorization of postulated initiating events (covering AOOs, DBAs and DEC/BDBAs)?

In accordance with requirements of the regulatory documents of the Russian Federation in the field of atomic energy use (federal rules and regulations) safety analysis of the NPP unit (including low power NPP) shall be performed for all plant states and consider all locations containing nuclear materials, radioactive substances and radioactive waste where abnormal operational conditions or accidents can occur.

In the national regulatory framework, specifically in the NP-001-15[43] 'General Provisions for Nuclear Power Plant Safety Assurance' along with the normal operation of the NPP unit, abnormal operational states are defined, which include anticipated operational occurrences, DBA and BDBA, including severe accidents.

It shall be noted that BDBA are going outside the framework of DBA, which substitute the basis for safety system design. BDBA can (not always) have heavier consequences than DBA (e.g. core meltdown). For such accidents the safety analysis shall utilize not conservative, but realistic approach, that is, an approach in which assumptions that lead to inherently more unfavorable results of the analysis are not intentionally made, and modeling of processes, as far as possible, is performed with the goal of reflecting its realistic behavior.

Integrated approach to formation of initiating event list and BDBA list for all NPP units with different types of reactors including low power NPP units with small modular reactors is established by requirements of NP-001-15[43] .

As such, in accordance with requirements of para. 1.2.14 of NP-001-15[43] , list of initiating events, which is developed by designer organizations of reactors and NPPs, shall include all possible external and internal hazards, which disrupt normal operation of NPP and are not excluded by inherent safety features of the reactor unit. Combinations of independent NPP systems (components) failures, personnel mistakes, internal and external hazards are accounted for in the aforementioned initiating events list only if it is explicitly required by the FRR requirements.

The NPP safety targets are the following:

- non-exceedance of cumulative severe accident probability for each NPP power unit within a year equal to 10^{-5} ;
- non-exceedance of cumulative probability of a large-scale emergency release for each NPP power unit within a year equal to 10^{-7} ;

- non-exceedance of cumulative severe accident probability for nuclear fuel storage facilities at the NPP (not included into the NPP power units) within a year equal to 10^{-5} .

However, it is allowed not to include internal hazards with assessed probability of occurrence lower than 10^{-6} per year or lower in the initiating events list. Moreover, the failures of the components (systems including them) are allowed not to be included if sufficiently high level of reliability is justified or during the period when the component (system) is out of operation for an established time for maintenance and repair. In addition, initiating events related to ruptures of equipment casings and vessels manufactured and operated in accordance with the most stringent quality requirements established by the federal rules and regulations in the field of atomic energy use also may be excluded from list of initiating events.

In accordance with established practice, requirements for the content of safety analysis reports for NPP units NP-006-16[44] (for VVER) and NP-018-05[47] (for fast reactors) include approximate lists of initiating events for DBA and approximate lists of BDBA which shall be used as a basis for development of definitive lists by the reactor unit designer organization and NPP designer organization that consider specific features of each particular unit. It shall be mentioned that approximate lists of initiating events and BDBA established in FRR were formulated after the beginning of operation of first-of-a-kind reactor unit and consider operational experience.

Approximate list of initiating events for analysis of design basis accidents with VVER reactors according to Appendix 7 NP-006-16[44] :

1. Initiating events with changes of reactivity and energy emission distribution:

- ejection of CPS CR;
- uncontrolled withdrawal of the working CPS CR group from the reactor core with the operating rate;
- misalignment of the CPS CR (falling of CPS CR, erroneous withdrawal of CPS CR, static misalignment of CPS CR);
- connection of a non-operating loop of the reactor coolant pipeline without prior power reduction;
- Inadvertent dilution of boric acid in the primary circuit coolant;
- error in the course of fuel assembly loading into the reactor;
- human error in the course of xenon oscillations suppression (movement of CPS CR resulting in the maximum possible deformation of the power density field).

2. Initiating events resulting in reduction of heat removal from the primary circuit:

- reduction of steam flow rate to the turbine (due to improper operation or failure of the steam pressure controller);
- closing of the turbine stop valves or loss of external electrical load;
- loss of flow from feedwater pumps;
- feedwater pipeline break;
- unauthorized closure of the main steam isolation valve;

- loss of vacuum in the turbine condenser or any other failures resulting in the turbine trip;
 - loss of auxiliary power supply of the NPP.
3. Initiating events resulting in increase of heat removal from the primary circuit:
- deviations in operation of the feedwater system causing decrease of feedwater temperature;
 - deviations in operation of the feedwater system causing increase of feedwater flow rate;
 - unintended opening of the secondary circuit steam relief valve with subsequent failure to seat;
 - increase of steam consumption per turbine (due to improper operation or failure of the steam pressure controller);
 - a range of steam pipeline ruptures inside and outside the containment.
4. Initiating events resulting in decrease of coolant flow rate in the primary circuit:
- disconnection of various number of RCPs for different initial states of the RF;
 - instant seizure of an RCP;
 - RCP shaft breakage;
 - emergency frequency deviation in the NPP auxiliary power supply network.
5. Initiating events resulting in decrease of coolant mass in the primary circuit:
- unauthorized opening of the pressurizer safety valve with subsequent failure to seat;
 - compensated primary circuit leakage inside the containment;
 - a range of primary circuit coolant leaks including gross-section rupture of the reactor coolant pipeline;
 - coolant leakages from the reactor during shutdown with unsealed reactor and in the course of refueling;
 - ruptures of pipelines connected to the primary circuit and crossing the containment resulting in leakages outside the containment;
 - breakage of a steam generator heat exchange tube;
 - breakaway of the steam generator header cover;
 - SG header breakage.
6. Initiating events caused by unauthorized actuation of safety systems:
- unauthorized actuation of the algorithm for management of a primary-to-secondary leakage accident;
 - unauthorized injection to the pressurizer;
 - unauthorized actuation of the ECCS;
 - unauthorized actuation of the SG emergency cooldown system;
 - unauthorized actuation of the passive heat removal system.
7. Initiating events caused by any occurrences in the course of SNF storage and handling:

- falling of separate fuel assemblies, leak-tight bottles, casings with fuel assemblies in the course of transportation and handling operations;
- falling of any objects (into the reactor or spent fuel pool) that can change the position and damage the integrity of fuel assemblies (in the reactor and spent fuel pool);
- breakage of packaging fasteners in the course of FA transportation;
- falling of transportation containers with spent fuel assemblies;
- failures of any components of the NF storage and handling systems;
- hang-up of a spent fuel assembly in the course of refueling operations;
- leakage from the fuel pool or rupture of a pipeline resulting in water level decrease in the fuel pool;
- failures in the SFP cooling system;
- reduction of homogeneous poison concentration in the SFP water;
- possible errors in the course of the core loading (refueling).

8. Initiating events caused by releases of radioactive media from auxiliary systems and equipment:

- medium leaks through the equipment seals;
- pipeline leakages in the RW transportation, storage and processing systems containing radioactive gas;
- leakage or damage of systems containing liquid radioactive media;
- medium leakage from any vessel containing radioactive substances.

9. Fires

10. Flooding

11. External impacts of natural and human-induced nature

Approximate list of initiating events for analysis of design basis accidents with fast reactors with liquid-metal sodium coolant according to para. 15.2.1 of NP-018-05[45] :

- flow passage reduction or blockage in one fuel assembly due to swelling of the materials, ingress of coolant impurities or foreign objects with subsequent breakage and melting of fuel elements;
- loss of leak-tightness in the primary circuit pipeline at any section without safety containment;
- loss of leak-tightness in the primary circuit gas system;
- loss of integrity in the spent fuel assembly drum.

Requirements that specify an approach to development of definitive lists of initiating events for DBA and definitive lists of BDBA are established in NP-006-16[44] (for VVER) and NP-018-05[45] (for fast reactors). Amongst other requirements it is stipulated that definitive lists of initiating events for DBA and definitive lists for BDBA shall be developed with consideration of:

- analysis of NPP components' failures, which were selected for further detailed analysis;
- operational experience of specified NPP unit and operational experience of analogue NPP units;
- probabilistic assessments.
- In addition, it shall be justified that definitive lists for accident analysis consider:
 - all types of events that could lead to accidents, namely, equipment failures, personnel mistakes, natural and human-induced external hazards (from sources located both on and off the NPP site), fires and explosions;
 - all possible locations of nuclear materials, radioactive waste and radioactive substances where an accident can occur: reactor unit, spent fuel pool, nuclear fuel storage, locations where nuclear materials could be during transportation process, containers and pipelines of radioactive waste management system and others;
 - all possible operational plant states (full power operation, low power operation, operation on minimum controllable power level, shutdown states for repair and refueling, «cold» state, «hot» state and other states, defined in NPP's design).

Therewith, consideration of representative scenarios of severe accidents with damage of fuel rods exceeding the maximum design damage limit should be justified for the definitive list of BDBA.

In accordance with requirements of para. 1.2.12 of NP-001-15[43] the established limits for design basis accidents shall not be exceeded in case of any initiating event considered in the NPP design in combination with a failure of any of the following safety system components independent of the initiating event in accordance with the single failure principle: an active component or a passive component with moving mechanical parts, or a passive component without any moving parts with the probability of failure to perform the safety function of 10^{-3} or more, or any human error independent of the initiating event.

Final lists of beyond design basis accidents (including severe ones) shall be presented in the NPP SAR. They shall include representative scenarios in order to define measures for management of such accidents. Representativeness of scenarios shall be provided by consideration of the NPP criticality level as well as potential operability or inoperability of safety systems and special-purpose engineering features for beyond design basis accident management.

Realistic (non-conservative) analysis of the above-mentioned beyond design basis accidents with assessment of development probabilities and consequences of beyond design basis accidents shall be given in the NPP SAR.

Programs used for such analysis must be qualified according to 'Review Procedure for Computer Programs Used to Build Calculation Models for Processes Effecting

the Safety of Nuclear Facilities and (or) Activities in the Field of Atomic Energy Use'. The procedure includes:

- Preliminary review
- Computer program testing. When testing a computer program, the customer of the examination demonstrates to the TSO the possibility of using the computer program by running the computer program by inputting the input data contained in the documents specified in subparagraph 'c' of para. 5 of this Procedure and performing calculations in accordance with the input data; comparing the results obtained in the process of performing the calculations with the results presented in the Report received from the customer of the expert review; calculation of the hash function for the main files of the received computer program. Analysis and development of a report on the results of the Analysis. Preparation of a draft certification passport for a computer program. The analysis evaluates:
 - the purpose of the computer program;
 - structure of the computer program, composition of its components (if any);
 - mathematical methods of building computational models of processes and phenomena affecting the safety of NPP, adopted in the computer program;
 - justification of applicability of used databases, libraries of constants, including justification of applicability of reference data on physical constants and properties of substances and materials;
 - justification of the applicability of the experimental data used for validation purposes of the computer program (justification of adequacy of reproduction by the computer program of a physical phenomenon and (or) process (justification of adequacy of reproduction by the computer program of the physical phenomenon and (or) process (or a set of phenomena and (or) processes) with an assessment of the error of the calculation result, performed with the help of the computer program, carried out by comparing the specified results of calculations with experimental data);
 - justification of the methodology for estimation of errors and uncertainties of calculation results obtained using the computer program, including justification of the ranges of parameters of calculation models for which the errors of the said calculation results are determined;
 - compliance of the Report with the structure and content specified in Annex No. 2 to this Procedure.

This approach follows the requirements of p.5 SSG-2, although in Russia the computer program review is supposed to be made one time. Subsequently, a developer can use the program in the scope of application without any additional qualification. To justify fulfillment of p.1.2.9 NP-001-15[43] the program passport must be specified in NPP SAR.

Analysis of beyond design basis accidents presented in the NPP SAR shall serve as the basis for the development of action plans in order to protect the personnel and the public in case of any accidents and also for the development of beyond design basis accident management guidelines. In accordance with requirements of para. 2.1.8 of NP-082-07 [47] [46] designers of reactor unit and NPP classify DBA and BDBA in accordance with frequency of occurrence and severity of the consequences. It should be justified in the reactor unit design and NPP design that the accepted design limits and safety criteria are assigned with consideration of the probability of occurrence of the initial event, more frequent initial events should be assigned stricter design limits and safety criteria.

For example, for VVER type reactors calculations of the radiological consequences of design basis accidents at values of initial data and model parameters, which provide for obtaining the highest values of predicted radiation doses to the critical group of population at the sanitary protection zone (SPZ) boundary and beyond, have shown fulfilment of the requirements of scientific and technical standards.

The calculations of radiological consequences of design basis accidents with the values of initial data and model parameters that provide the highest values of predicted doses to the critical group of population at the SPZ boundary and beyond its boundaries, have shown the fulfilment of the requirements of scientific and technical documentation of the Russian Federation SP AS-03[49] and NRB99/2009[50] .

It was shown that:

- at the early stage of the accident (during the first ten days) the requirements of the SP AS-03[49] for new NPP projects, i.e. levels A of protective measures (sheltering, iodine prophylaxis and evacuation) are not exceeded, iodine prophylaxis and evacuation) established in NRB-99/2009[50] ;
- calculated values of annual effective doses to the population do not exceed the limits of public exposure doses established in the ToR for the development of the construction project for design basis accidents of category 3 (1 mSv/event) and for design basis accidents of category 4 (5 mSv/event).

For beyond design basis accidents radiation consequences were calculated for all modes. The results obtained from calculation of radiation doses to the population in beyond design basis accidents without fuel melting showed that:

- in all modes the requirements of SP AS-03[49] established for the newly designed NPPs, according to which the consequences of accidental releases should not lead to radiation doses to the population in case of beyond design basis accidents without fuel meltdown.
- should not lead to doses of radiation exposure of the population, which require taking any protective measures in the initial period of the accident, i.e. level A should not be exceeded, that according to NRB-99/2009[50] corresponds to a dose of 5 mGy to the whole body and 50 mGy to the thyroid gland;
- no resettlement of the population at distances exceeding 250 m from the reactor.

Requirements for identification and categorization of postulated initiating events for BDBA are presented in p.2.5 RB-150-18[48] According to para 2.5.2 of the document:

The set of initial accident events includes the following:

- internal events (events related to failures of plant elements, including failures of plant elements due to common causes, as well as to personnel errors);
- events related to on-site fires, on-site flooding;
- events related to external impacts of natural and man-made nature that meet the following conditions:
 - the event disrupts the normal operation of the nuclear plant;
 - the event requires a response on the part of the systems (elements) of the plant and (or) plant personnel to prevent the event from turning into an accident, or directly leads to an accident.

To determine the internal events to be included to the postulated initiating events list the following methods are recommended for joint use

- Analysis of failures and their consequences for the systems and elements of the nuclear plant. In the course of this analysis, system by system, the impact of various types of failures of elements or groups of elements included in the system on the normal operation of the plant is qualitatively analysed (failures during operation of the plant elements are analysed, including failures of the spontaneous operation type, and also failures of the including spontaneous operation failures, as well as demand failures, including opening and closing failures, are analysed, including failures to open and close, failures to change position, failures to start-up. Common cause failures are also analysed);
- use of initiating events lists from previously performed safety analyses, including from WABs for the analysed and similar NPP units, as well as from generalized lists of IEs from authoritative foreign and international sources;

In order to define on-site fire (on-site flooding) events to be included in the set of initial accident events, it is recommended that:

- Fire zones (inundation zones) be allocated in such a way that each of them corresponds to the definition of a fire zone (inundation zone) presented in the Appendix No. 2 to this Safety Guide. It is not permissible for one designated fire zone (flood zone) to include premises that also belong to another fire zone). It is also not permitted that outside the defined set of fire zones (flood zones), there should remain systems and elements important for safety;
- postulate the occurrence of fire (flooding) alternately in each of the allocated zones and analyze the impact of such a fire (flooding) on the safety of the nuclear plant (assuming that all equipment (and flooding) in the fire zone (flooding zone) has been removed);
- assuming that all equipment (other elements of the plant) located in the corresponding zone fails).

On-site fires (on-site flooding) in those fire zones (flooding zones) that fulfil the requirements specified in para. 2.5.2 of this Guide shall be included in the list of initiating events list.

For the determination of natural and man-made external initial events, to be included in the list of initial accident events, it is recommended:

- for each of the external factors (processes, phenomena) of natural and man-made origin the sources of external impacts corresponding to these factors (processes, phenomena) should be analysed at the site and in the area where the plant is located, capable of having an impact on the plant that meets the conditions specified in para. 2.5.2 of the Safety Guide. External impacts, satisfying these conditions are included in the set of initial accident events;
- Analyze the possibility of a combined adverse effect of a combination of several external factors (processes, phenomena), as well as the occurrence of additional processes, phenomena and factors, identifying those cases where the combined effect of factors (processes, phenomena) leads to more severe consequences than the effects of each of the factors separately, to include the identified combinations in the set of initial accident events, if they fulfil the conditions specified in para. 2.5.2 of this Safety Guide;
- if for one and the same external factor (process, phenomenon) of natural (man-made) origin the scale of impact on the plant and the scope of measures required from the plant systems and personnel to ensure plant safety are different at different intensities of the factor (process, phenomenon) under consideration, then external impacts corresponding to one and the same factor (process, phenomenon) are included as separate initial events in the set of initial accident events.

Impacts corresponding to the same factor (process, phenomenon), but having different intensities, thus causing different in importance disturbances of normal operation of the plant and requiring different reactions of systems and operators of the plant.

Annex to answer question 3

Anticipated operational occurrence at the NPP - any disturbance in the NPP operation with deviation from the established operation limits and (or) conditions. In this case other limits and (or) conditions established in the NPP design particularly safe operation limits and (or) conditions may be also exceeded.

Design basis accident - an accident with the initiating events and end states defined in the NPP design and the provided safety systems aimed to assure mitigation of its consequences to the limits established for such accidents in case of any failure of a safety system component independent from the initiating event and considered in the NPP design or in case of a human error independent from the initiating event.

Beyond design basis accident - an accident caused by initiating events not considered for design basis accidents or accompanied by additional failures of safety system components as compared to design basis accidents in excess of a single failure and implementation of erroneous personnel decisions.

Severe accident - a beyond design basis accident with damage to fuel elements exceeding the maximum design limit.

Maximum design limit - allowed values of fuel element parameters and characteristics in conditions of design basis accidents, exceeding which may lead to the destruction of fuel elements.

Large-scale emergency release - release of radioactive substances into the environment in case of any accident at the NPP when it is necessary to implement measures for protection of the public within the initial stage of the accident at the border of the protective actions planning zone established in accordance with the requirements of NPP siting regulations and rules and outside it.

4. **Would your existing requirements for identification and categorization of PIEs - postulated initiating events (covering AOOs, DBAs and DEC/BDBAs) be applicable to reviews of SMRs including Gen-IV SMRs? If not, why and how would you go about it? Please describe, in your professional opinion (does not have to represent the organizational 'consensus'), what thinking/method would you use to assess PIEs?**

Requirements of the national regulatory documents in the field of atomic energy use, namely NP-001-15[43] and NP-082-07[47], regarding identification and categorization of alleged anticipated operational occurrences are neutral and fully applicable to small power NPP with SMR. At the same time, as it was noted earlier, definitive lists of DBA and BDBA (including severe accidents) for small power NPP with SMR currently are not defined in the national regulatory framework in the field of atomic energy use.

Taking into account the established practice of development of definitive lists of anticipated operational occurrences and accidents by the developers of reactor unit and NPP on the basis of approximate lists established by requirements for the content of safety analysis reports for NPP units NP-006-16 [44] (for VVER) and NP-018-05[45] (for fast reactors), it is practical to provide for development of approximate lists of initiating events at the early stages with participation of reactor unit and NPP designers that will consider the operational experience of analogue nuclear facilities (e.g. research reactors), with their subsequent inclusion in the requirements for the content of safety analysis reports for NPP units. At the same time, in accordance with the requirements of the NP-001-15[43] the designers of the low power NPP with SMR could exclude external or internal events from the list of initiating events of DBA which are eliminated by inherent safety features of SMR and Gen IV reactors and their design principles.

5. **Does your country use a source term only covering specific radionuclides (source term 'list')? If yes, is the list adequate and relevant to SMRs including Gen-IV SMRs?**

The regulatory documents of the Russian Federation (federal rules and regulations in the field of atomic energy use, sanitary norms and rules) do not contain a specific list of radionuclides that should be considered at source term assessment.

The most common and hazardous radionuclides such as ^{131}I , ^{134}Cs , ^{137}Cs , ^{90}Sr , ^{238}Pu , ^{239}Pu , ^{241}Am according to NRB-99/2009[50] are considered in any case. The other

radionuclides vital for the evaluation model are being selected at the discretion of the designer and according to indirect requirements and general safety considerations. This approach is more flexible and applicable to all nuclear energy facilities, including SMRs.

6. In your country, how are the quantitative safety goals on releases (typically I-131 and Cs-137 are the radionuclides used in quantitative safety goals) defined? Would these quantitative safety goals be applicable to SMRs including Gen-IV SMRs? If not, what would be the regulatory process for an exemption?

The regulatory documents of the Russian Federation (federal rules and regulations in the field of atomic energy use, sanitary norms and rules) do not define specific numerical safety values for the source term. Instead, safety criteria are used based on potential radiation doses to site personnel and the public. This principle is enshrined in the following normative documents. Para. 1.2.1 of NP-001-15[43] states that:

“An NPP complies with the safety requirements if the following conditions are met: exposure of the personnel, the public and the environment to the NPP radiation under normal operation and normal operation malfunctions (including design basis accidents) does not exceed any established exposure limits for the personnel and the public as well as any release and discharge standards; exposure of the personnel, the public and the environment to NPP radiation in a beyond design basis accident is limited.”

NRB-99/2009[50] – ‘Requirements for limiting anthropogenic exposure under controlled conditions’

Two classes of standards are established for categories of exposed persons:

- basic dose limits (PD)
- permissible levels of monofactor exposure (for one radionuclide, pathway or one type of external exposure), which are derived from the basic dose limits: annual intake limits (AIs), permissible annual average volumetric activity (AAV), annual average specific activity (ASA) and others.

To ensure conditions under which radiation exposure will be below the permissible level, considering the level of radiation safety achieved in the organization, the administration of the organization additionally establishes control levels (doses, activity levels, flux densities, etc.).

The effective dose for personnel shall not exceed for the period of labour activity (50 years) - 1000 mSv, and for the population for the period of life (70 years) - 70 mSv.

This approach is more flexible and applicable to all nuclear energy facilities, including SMRs.

APPENDIX B.8 - SOUTH AFRICA**National Nuclear Regulator (NNR) response to the Questionnaire on Mechanistic Source Term****1. Do you have regulatory requirements pertaining to mechanistic source terms?**

No. The NNR does not have specific requirements pertaining to mechanistic source terms. The NNR's regulations on source terms are rather generic and with the right interpretation can be applied to mechanistic source terms. These regulations can be found in the soon to be updated Specific Nuclear Safety Regulations for nuclear facilities (SNSR), Regulations on licensing of sites for new Nuclear Power Plants (No.R.927)[51] , RD-0024[52] , RG-0019[53] and other documents. It is worth noting that RG-0019 goes into more detail on the assessment of source terms and has more potential for application to mechanistic source terms.

2. If not, what existing regulatory requirements could be used in your reviews of mechanistic source terms?

The NNR applies regulatory requirements on the applicant or licensee's safety related programs & processes, which require the applicant or licensee to demonstrate safety, compliance with regulatory safety criteria & requirements as well as demonstrate good engineering practices and the use of codes and standards. The NNR requires the licensee or applicant to provide information regarding the type of nuclear installation, potential source terms, barrier concept/s with respect to confinement of fission, activation and/or radioactive products, heat removal, reactivity control etc.

3. Do you have requirements for identification and categorization of postulated initiating events (covering AOOs, DBAs and DEC/BDBAs)?

Yes. These are currently documented in RG-0019 and the draft Specific Nuclear Safety Regulations: Nuclear Facilities. The NNR regulatory documents stipulate that the concept of DiD are to be applied to all safety related activities. Deterministic safety analysis should be used to assess the adequacy of the design and should cover both normal operations and accident conditions (Design Basis and DEC) and should be supported by appropriate probabilistic analysis to judge the significance of uncertainties, show that risks are balanced, and demonstrate compliance with numerical risk criteria. Deterministic safety analysis should be used to analyze AOO's, DBA's and DBEC's.

The NNR requires that PSA must be performed to demonstrate compliance with the numerical risk criteria unless it can be justified that no credible accident conditions exist.

Nuclear Safety Criteria is stipulated further in regulatory documents SSRP, RG-0019 and RD-0024.

4. Would your existing requirements for identification and categorization of PIEs - postulated initiating events (covering AOOs, DBAs and DEC/BDBAs) be applicable

to reviews of SMRs including Gen-IV SMRs? If not, why and how would you go about it? Please describe, in your professional opinion (does not have to represent the organizational 'consensus'), what thinking/method would you use to assess PIEs?

Yes, the NNR draft regulations have been reviewed for applicability to SMR technologies and certain changes have been proposed. IE, PIE's and LBE must be defined by the applicant designer using a systematic approach. These are typically technology-dependent. Currently the NNR views the application of requirements to large scale PWRs are the same as would be applied to SMRs.

5. Does your country use a source term only covering specific radionuclides (source term 'list')? If yes, is the list adequate and relevant to SMRs including Gen-IV SMRs?

The licensee considers a source term for a large number of isotopes and must justify through research and development the applicable list of isotopes to be considered. New radionuclides that may arise due to different fuel/coolant composition or unexplored release pathways can still be appended on the source term list, depending on whether their dosage amount falls in the inclusion criteria of the source term list. For this reason, the NNR approach to defining the source term list is adequate to SMRs including GenIV SMRs.

6. In your country, how are the quantitative safety goals on releases (typically I-131 and Cs-137 are the radionuclides used in quantitative safety goals) defined? Would these quantitative safety goals be applicable to SMRs including Gen-IV SMRs? If not, what would be the regulatory process for an exemption?

The NNR regulatory framework includes principal safety criteria on dose and risk which requires the applicant or licensee to demonstrate the derivation of target release quantities, the so called Annual Authorized Discharge Quantities (AADQ's), for NO's and possible AOO's which are required when establishing the design basis parameters for the design of nuclear installations.

The ALARA principle needs to be further applied during design and operation. According to section 4.5.2.1 of No.R.388, "where applicable in terms of the prior safety assessment, the optimization of radiation protection must be subject to dose constraints specific to the authorized action, which must not exceed values that can cause the relevant dose limits to be exceeded and which will ensure as far as practicable that doses are restricted by application of the ALARA principle on a source-specific basis rather than by dose limits."

APPENDIX B.9 - UNITED KINGDOM**Office for Nuclear Regulation (ONR) response to the Questionnaire on Mechanistic Source Term****1. Do you have regulatory requirements pertaining to mechanistic source terms?**

ONR is a goal setting regulatory body and does not have any specific requirements relating for a duty holder to perform analysis of mechanistic source terms. ONR's guidance to inspectors set expectations. These take the form of the ONR's safety assessment principles and technical assessment guides¹⁵.

Within the guidance there are several aspects of ONR's regulatory guidance that address this topic area.

- FA.15 sets the expectation that for each state or scenario, the SAA should determine the magnitude and characteristics of the predicted source term and its potential radiological consequences, including societal effects.
- AV.2 sets the expectation that calculation methods used for the analyses should adequately represent the physical and chemical processes taking place
- NS-TAST-GD-030 - sets the expectation that, for source terms:
 - The modelling method/s used to perform source term analysis are clear.
 - The radionuclide grouping scheme used for the source term analysis is consistent with current state-of-the-art practice.
 - All the assumptions made to obtain source terms are described and justified.
 - The computer code calculations used as the basis for estimating facility-specific source terms for selected accident sequences are documented.
 - If there are cases where facility-specific computer code calculations were not performed, the method by which source terms have been estimated is described and justified. Also, the relationship between the deterministic accident progression analyses and deterministic source term analyses are clearly described and justified.

ONR also expect that Level 3 PSA is performed, and that the results are compared against appropriate radiological targets. ONR therefore expect that the source term derived in the Level 2 PSA is in the form that it can be used for Level 3 PSA (NS-TAST-GD-030).

2. If not, what existing regulatory requirements could be used in your reviews of mechanistic source terms?

See response to Q1.

¹⁵ <https://www.onr.org.uk/guidance.htm>

3. Do you have requirements for identification and categorization of postulated initiating events (covering AOOs, DBAs and DEC/BDBAs)?

ONR expects that a systematic, auditable and comprehensive approach is taken to identify all faults with the potential to lead to effective doses greater 0.1 mSv to a worker or 0.01 mSv to a member of the public (SAPs FA.2). ONR's guidance in NS-TAST-GD-006 provides guidance as to typical techniques that are expected by ONR, but they are not prescribed.

ONR SAPs FA.4 to FA.9 set the expectations for the treatment of design basis faults. ONR does not differentiate between AOOs and design basis accidents, although a differentiation is made between frequent ($> 10^{-3}$ pa) and infrequent ($10^{-3} > f > 10^{-5}$ pa).

ONR expects that design basis analysis is performed on all faults that have the potential to lead to unmitigated radiological doses greater than those set out in SAP NT.2 Target 4. Target 4 takes a graded approach, and the cut-off frequency for consideration of design basis analysis decreases by one order of magnitude as the unmitigated effective dose increases. Details of the derivation of this region can be found in ONR's SAPs.

FA.7 states that design basis techniques should be applied to sequences down to a cut-off frequency of 10^{-7} pa. An implication of this is that initiating events with common cause failure of the design basis safety measures should also be treated within the design basis. This differs slightly to the SSG-2, which states that sequences with multiple failures should be treated as DEC-A sequences. However, ONR recognizes that conservatism in the approach may be relaxed as the sequence frequency decreases.

ONR's SAPs have not adopted the design extension condition categorization. However:

For beyond design basis accidents (but not severe accidents), it is ONR's expectation that it should be demonstrated that a small change in a DBA parameter will not lead to a disproportionate increase in radiological consequences, i.e. there should be no cliff edge effect.

For severe accidents, ONR expects that severe accident analysis is performed on accidents with off-site consequences with the potential to exceed 100 mSv, or to a substantial unintended relocation of radioactive material within the facility that places a demand on the integrity of the remaining physical barriers (FA.15).

4. Would your existing requirements for identification and categorization of PIEs - postulated initiating events (covering AOOs, DBAs and DEC/BDBAs) be applicable to reviews of SMRs including Gen-IV SMRs? If not, why and how would you go about it? Please describe, in your professional opinion (does not have to represent the organizational 'consensus'), what thinking/method would you use to assess PIEs?

Yes - as stated above, ONR expects comprehensive identification of initiating faults as part of the design basis and beyond design basis analysis. ONR's expectations

are technology neutral, and ONR do not prescribe a list of initiating events to be considered. ONR's NS-TAST-GD-006 sets expectations for the types of fault identification processes that it considers to be relevant good practice.

It should be noted that ONR's numerical target 4 applies down to even relatively low unmitigated effective doses.

In addition, since ONR's definition of a severe accident is related to either effective doses or relocation of radioactive material, ONR's expectations for severe accident analysis would apply (FA.15), even for facilities in which core melt was not deemed credible.

5. Does your country use a source term only covering specific radionuclides (source term 'list')? If yes, is the list adequate and relevant to SMRs including Gen-IV SMRs?

No, ONR does not specify a list of radionuclides to be considered in the source term analysis.

6. In your country, how are the quantitative safety goals on releases (typically I-131 and Cs-137 are the radionuclides used in quantitative safety goals) defined? Would these quantitative safety goals be applicable to SMRs including Gen-IV SMRs? If not, what would be the regulatory process for an exemption?

ONR has no safety goals for radiological releases. No specific targets relating to core damage frequency, large release frequency and large early release frequency are defined. No specific targets are defined for practical elimination of large or early releases. However, ONR do set expectations for effective doses which are technology neutral.

ONR's SAPs set the expectations for numerical targets (NT.2). ONR differentiate between Basic Safety Objectives (BSO) and Basic Safety Levels (BSLs). Besides BSL for normal operation, which is a legal limit, the BSOs and BSLs are goal setting. The BSO is set lower than the BSL. Whilst ONR considers that the As Low As Reasonably Practicable (ALARP) concept applies regardless of whether the BSO is achieved, ONR uses the BSO as an indicator for judging whether risks have been reduced ALARP. In general, ONR considers anything higher than the BSL to be intolerable and is unlikely to meet the expectations that risks have been reduced ALARP, particularly for new facilities.

The targets relating to accident conditions can be summarized as:

- Target 4 - frequency targets for dose bands relating to design basis accidents, assessed conservatively
- Target 5 - individual risk of death from accidents - any person on the site
- Target 6 - BSO and Frequency dose targets for any single accident - any person on the site
- Target 7 - individual risk to people off the site from accidents (summed risk of all accidents)

- Target 8 - Frequency dose targets for accidents on an individual facility - any person off the site
- Target 9 - Societal risk - Total risk of 100 or more fatalities

In general, these targets would be equally applicable to Gen-IV SMRs. Any specificities of a reactor design which may result in the BSOs/BSLs being challenged would be considered when judging whether risks have been reduced ALARP. ONR's guidance on the ALARP principle is laid out in NS-TASG-GD-005.

APPENDIX B.10 - UNITED STATES**U.S. Nuclear Regulatory Commission (U.S. NRC) response to the Questionnaire on Mechanistic Source Term****1. Do you have regulatory requirements pertaining to mechanistic source terms?**

U.S. regulations currently assume a deterministic design basis analysis; however, the response to question 2 provides additional context.

2. If not, what existing regulatory requirements could be used in your reviews of mechanistic source terms?

Title 10 of the Code of Federal Regulations (10 CFR) Parts 50, 52, and 100 give regulatory requirements related to licensing of power reactors.

While MST is not defined or described in regulations; 10 CFR 50.2 defines that "source term refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release."

The regulatory requirements listed below give offsite dose reference values and describe the features of the analysis to be used in design basis assessment of accidents for new reactor applications:

- 10 CFR 50.34(a)(1)(ii)(D) requires a safety assessment in the preliminary safety analysis report of

The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release³ from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that:

- *An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).*
- *An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from*

the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).

With respect to the source term, footnote 3 states:

The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

- 10 CFR 50.34(a)(1) is referenced by the power reactor siting requirements in 10 CFR 100.21.
- Requirements analogous to 10 CFR 50.34 are given in 10 CFR Part 52 for the safety analysis reports for power reactor applications for early site permits (10 CFR 52.17(a)(1)(ix)), standard design certifications (10 CFR 52.47(a)(2)(iv)), combined licenses (10 CFR 52.79(a)(1)(vi)), standard design approvals (10 CFR 52.137(a)(2)(iv)), and manufacturing licenses (10 CFR 52.157(d)).
- 10 CFR 50.67 provides for currently operating reactors to voluntarily adopt an alternative source term and has effectively similar requirements.

Use of an MST in the design basis safety analysis may require the applicant to request exemption from certain portions of the above regulations. Requirements for requesting exemptions are described in 10 CFR 50.12.

SECY-93-092 'Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements' describes an MST as "the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs." In this paper, the NRC staff recommended that source terms for non-LWRs be based upon a mechanistic analysis and that the acceptability of an applicant's analysis will rely on the NRC staff's assurance that the following conditions are met:

- The performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on the reactor and fuel performance through the research, development, and testing programs to provide adequate confidence in the mechanistic approach.
- The transport of fission products can be adequately modeled for all barriers and pathways to the environs, including the specific consideration of containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured.

- The events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties.
- The design-specific source terms for each accident category would constitute one component for evaluating the acceptability of the design.

These discussions of MST occurred under the context of design basis accident assessment. Normal operational effluent release analysis has not been included in discussions of development of mechanistic source terms, although development of effluent source terms could similarly be based on mechanistic modeling of radionuclide release and transport phenomena.

More recently, the NRC staff issued Regulatory Guide (RG) 1.233 'Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors'. The NRC staff is also developing a set of regulations called 'Part 53' as an alternative regulatory framework for non-LWR designs and other new commercial nuclear plants. These efforts rely more heavily on non-LWR probabilistic risk assessment (PRA). Trial RG 1.247 'Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities' endorses, with staff exceptions, a consensus PRA standard and industry guidance on PRA peer review. ASME/ANS RA-S-1.4-2021 'Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants' includes a technical element on development of mechanistic source terms for non-LWR PRA.

The NRC public website¹⁶ includes additional information on nuclear power reactor source term.

3. Do you have requirements for identification and categorization of postulated initiating events (covering AOOs, DBAs and DEC/BDBAs)?

The NRC has requirements and related guidance for the identification and analysis of AOOs, postulated or design basis accidents (DBAs), beyond design basis events, and severe accidents. The existing regulations related to AOOs and DBAs, such as 10 CFR 50.34 and 10 CFR 50.46, have to date been met through traditional or deterministic approaches that focus on analyses of postulated initiating events (PIEs) and the performance of selected fission product barriers (e.g. fuel cladding, reactor coolant pressure boundary, containment structures). Additional information related to the identification and analysis of AOOs and DBAs for light-water reactors (LWRs) is provided in Chapter 15 of the NRC's Standard Review Plan (NUREG-0800). NRC regulations applicable to operating and new LWRs also specify the need to include design features and programmatic controls to reduce the likelihood or severity of selected beyond-design-basis events such as anticipated transients without scram (10 CFR 50.62), station blackout (10 CFR 50.63), and specified beyond-design-basis external events from natural phenomena (10 CFR 50.155).

¹⁶ <https://www.nrc.gov/reactors/new-reactors/advanced/references/nuclear-power-reactor-source-term.html>

Additional analyses are performed based on required containment performance standards and site characteristics to estimate offsite doses for comparison to established reference values. This analysis is based upon a major accident, which has generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products. Additional information on the source term for the analyses is provided in RG 1.183 'Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors'.

NRC policies and regulations for new reactors (licensed after 2006) require assessments and related design features to address severe accidents (NRC Severe Accident Policy Statement), the impact of a large commercial airliner (10 CFR 50.150), and insights from probabilistic risk assessments (10 CFR 50.71). The NRC's experience to date with the assessments of severe accident design features has been with LWRs and the design features have involved providing measures to maintain containment integrity and limit the radiological releases following a core melt accident.

As described in question 2, more recently, the NRC issued RG 1.233 to support an increased use of PRA to identify and categorize a range of licensing basis events, support the safety classification of structures, systems and components, and assess defense in depth. The methodology endorsed in RG 1.233 is technology inclusive and therefore uses offsite radiological consequences as performance measures instead of defining requirements for specific fission product barriers. The consideration of offsite consequences for a variety of reactor technologies includes the use of PRAs and mechanistic source terms for various event sequences as described in RG 1.247.

- 4. Would your existing requirements for identification and categorization of PIEs - postulated initiating events (covering AOOs, DBAs and DEC/BDBAs) be applicable to reviews of SMRs including Gen-IV SMRs? If not, why and how would you go about it? Please describe, in your professional opinion (does not have to represent the organizational 'consensus'), what thinking/method would you use to assess PIEs?**

The NRC issued application guidance, including the identification and analysis of a spectrum of PIEs, and is engaging with several prospective applicants for SMRs of different light-water and Gen-IV technologies. These activities involve the use of both methodologies described under question 3.

In general, prospective applicants for LWR SMRs are electing to use the traditional or deterministic methodology with some adjustments to reflect specific attributes of SMR designs as well as risk insights from PRAs. Under such approaches, risk insights are usually presented in terms of core damage and large releases and do not include detailed development of mechanistic source terms for various event sequences. In terms of the assumed source terms for use in complying with the requirements to assess containment performance and site characteristics, the prospective applicants for LWR designs may use the guidance in RG 1.183 with some adjustments to address specific design features.

Prospective applicants for Gen-IV SMRs have generally elected to use the approach described in RG 1.233, which involves the increased use of PRA and the development of mechanistic source terms for specific event sequences from the PRA for a design and site. Additional guidance on this approach is provided in RG 1.247 and RG 1.253 'Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors'. As previously discussed, the NRC is also undertaking a rule making ('Part 53') that may better support the use of the approach outlined in RG 1.233 than do the current regulations in Parts 50 and 52.

5. Does your country use a source term only covering specific radionuclides (source term 'list')? If yes, is the list adequate and relevant to SMRs including Gen-IV SMRs?

The NRC does not identify a required source term list covering specific radionuclides. The analysis source term should consider the inventory of fission products in the reactor and available for release based on the maximum full-power operation of the reactor and include those radionuclides that are significant to assessment of the dose to an individual from exposure to the radionuclide release from the facility.

As an example, NRC regulatory guidance on an acceptable light water reactor accident source term is given in RG 1.183. Table 6 lists the radionuclide groups and their associated elements that should be considered in the NRC-defined LWR source term for design basis analyses. Tables 3 and 4 provide steady-state fission product inventory in the fuel rod plenum and gap as fraction of the rod inventory for boiling and pressurized water reactors, respectively, and list I-131, I-132, and Kr-85 separately from other halogens, other noble gases, and alkali metals. The radionuclide group information is applicable to LWRs similar to the currently operating U.S. reactors and may not be fully applicable to non-LWRs or LWR designs that are very different. For example, activation products such as tritium and Ar-41 may be more important for salt-cooled reactors but are not included in the RG 1.183 LWR core release source term as being less important than large releases of fission products for evaluation of dose to an individual exposed to the airborne accident release plume at offsite locations. RGs are not NRC regulations, and compliance with them is not required. Methods and solutions that differ from those set forth in RGs are acceptable if supported by a basis for the issuance or continuance of a permit or license.

6. In your country, how are the quantitative safety goals on releases (typically I-131 and Cs-137 are the radionuclides used in quantitative safety goals) defined? Would these quantitative safety goals be applicable to SMRs including Gen-IV SMRs? If not, what would be the regulatory process for an exemption?

The NRC has not defined quantitative safety goals on releases in terms of amount of radionuclide released. Quantitative health objectives are defined instead in terms of risk of health effect (early fatality, latent cancer fatality).

The NRC's safety goals are technology-inclusive and, hence, apply to SMRs including Gen-IV SMRs.

The NRC's safety goals are not regulatory requirements, and failure to meet them does not require an exemption from existing regulations (i.e. the NRC does not use risk-based regulation). Rather, the safety goals provide one element of a risk-informed regulatory philosophy.

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ABBREVIATIONS

AOO	Anticipated Operational Occurrence
BDBA	Beyond Design Basis Accident
DBA	Design Basis Accident
DEC	Design extension conditions
DSA	Design and Safety Analysis
FMEA	Failure Mode and Effect Analysis
Gen-IV	Generation IV
HAZOP	Hazard and Operability Analysis
HTGR	High Temperature Gas Reactor
LWR	Light Water Reactor
MST	Mechanistic Source Term
NPP	Nuclear Power Plant
PIE	Postulated Initiating Event
PIRT	Phenomena Identification and Ranking Table
PSA	Probabilistic Safety Analysis
SMR	Small Modular Reactor
SMR RF	Small Modular Reactor Regulators' Forum
TMI	Three Mile Island
WG	Working Group

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