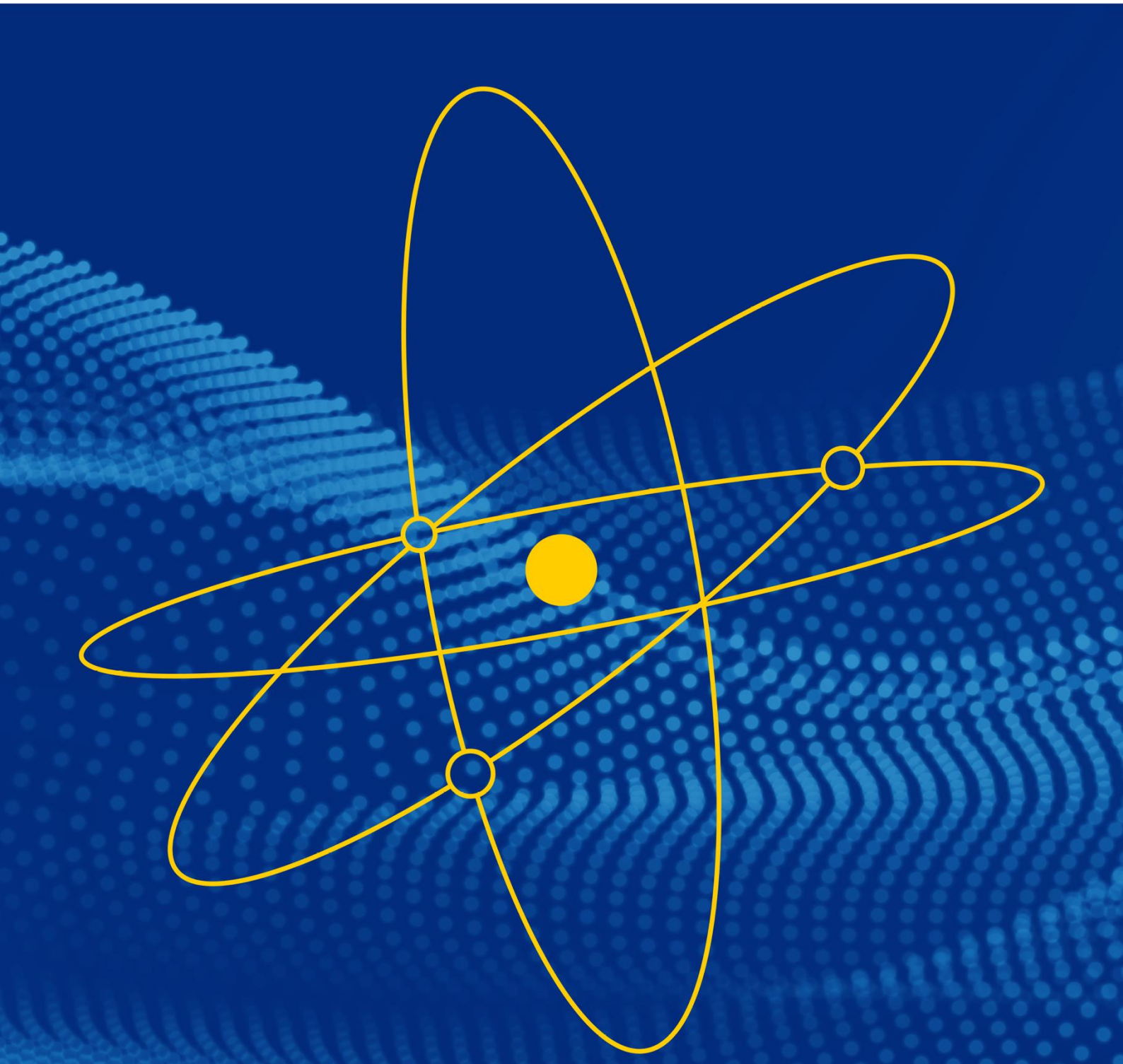
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Mechanistic Source Term Assessment for Generation-IV Systems



November 2026

GIF/RSWG/2026/002

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Abstract

The Risk and Safety Working Group (RSWG) has investigated source term assessment methodologies as well as how those methodologies relate to the concept of a “mechanistic source term.” A preliminary review determined that (1) “source term” is not used consistently and the meaning can differ based on the context of the specific licensing regime and its history, (2) the concept of a bounding accident source term does not align with the safety approach for Generation IV reactors, and (3) Generation IV reactors will have different source terms than most large, water-cooled reactors. This paper proposes a definition of “mechanistic source term” for Generation IV reactors to address these issues. Discussion is provided on the attributes of mechanistic source term assessment, including identification of events (accident sequences) for which the source term must be evaluated, prediction of the release of radionuclides from fuel during those events, and prediction of radionuclide transport, retention, and release to the environment using best-estimate phenomenological models. Technical considerations for the assessment of mechanistic source terms for Generation IV reactors are proposed.

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Acknowledgements

The authors would like to acknowledge the Design and Safety Analysis (DSA) Working Group of the Small Modular Reactor (SMR) Regulator's Forum. This work was informed by background information provided by the DSA Working Group as well as some preliminary discussions that took place between RSWG and DSA Working Group members. These exchanges were facilitated by the Scientific Secretary of the SMR Regulator's Forum.

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Figure 1: Concept of frequency-consequence target, reproduced from [4]. The practically eliminated situations (PES) with large or early releases are shown to be outside the plant design envelope. 6

List of Acronyms

AOO	Anticipated operational occurrence
DBA	Design basis accident
DEC	Design extension condition
EG	[GIF] Experts Group
EPR	Emergency preparedness and response
GIF	Generation IV International Forum
IAEA	International Atomic Energy Agency
LOCA	Loss-of-coolant accident
LWR	Light water reactor
MSR	Molten salt reactor
NO	Normal operation
RSWG	Risk and Safety Working Group
PES	Practically eliminated situation
PG	[GIF] Policy Group
SSC	System, structure, and component
VHTR	Very-high-temperature reactor

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1. Introduction

The Generation IV International Forum (GIF) Risk and Safety Working Group (RSWG) was formed to promote a consistent approach to safety, risk, and regulatory issues among Generation IV (Gen- IV) systems. The RSWG also advises the GIF Experts Group (EG) and Policy Group (PG) on matters such as:

- Gen-IV safety goals and evaluation methodologies to be considered in design and supporting research and development programs, and
- Interactions with the nuclear safety regulatory community, the International Atomic Energy Agency (IAEA), and relevant stakeholders.

The RSWG has produced several documents that describe frameworks and methodologies for the safety design and assessment of Gen-IV systems according to its mandate. These include:

- *An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems* [1]
- *Guidance Document for Integrated Safety Assessment Methodology (ISAM) – (GDI)* [2]
- *Basis for the Safety Approach for Design & Assessment of Generation IV Nuclear Systems* [3]
- *A Risk-Informed Framework for Safety Design of Generation IV Systems* [4]

Pursuant to the technology safety and reliability goals of GIF [5], which states that the Gen-IV reactors are to (1) excel in safety and reliability, (2) have a very low likelihood and degree of reactor core damage, and (3) eliminate the need for offsite emergency response, the GIF EG and PG also endorsed the RSWG to investigate existing source term assessment methodologies for these systems, as well as how those methodologies relate to the concept of a **mechanistic source term**.

In international safety standards, source term is defined as:

> "*The amount and isotopic composition of radioactive material released (or postulated to be released) from a facility*" [6].

Source term estimates are the quantified radiological releases to the environment from nuclear installations. Subsequent modelling of the transport of radioactive material in the environment, and the resulting health effects (dose) to the population, enable quantification of the consequences of events at nuclear facilities, including postulated accidents. Source terms are, therefore, a critical component of reactor safety assessment, siting, and emergency preparedness and response (EPR) considerations.

This paper describes what is meant by "mechanistic source term" as well as the role of mechanistic source term assessments in the safety design and demonstration of Gen-IV reactors.

2. Context and Motivation

In the prescriptive licensing of many Generation II light water reactors (LWR), "source term" was associated with a severe accident (severe core degradation) and the postulated release of a significant fraction of the core radionuclide inventory into containment [7]. These source terms were based on conservative assumptions of radionuclide releases that reflected the best available knowledge at the time. Although subsequent advances in the understanding of fuel damage, fission product release, and transport phenomena facilitated more realistic source

term estimates [8], the underlying conservative assumptions (in particular, severe core degradation) remained. The licensing bases for many LWRs continue to reference these source terms, typically when operating within more prescriptive regulatory regimes.

In some contexts, the source term was explicitly the release to the environment from the "maximum credible accident" or "maximum hypothetical accident" at the facility. This was assumed to be a large loss-of-coolant accident (LOCA) with core melting and intact containment for the early LWRs [7]. The release from the maximum credible accident was used to bound the requirements for siting and EPR, like the necessary size of the facility exclusion area and the minimum distance to population centers. This release was therefore referred to as the design basis accident (DBA)[1] source term in some jurisdictions [9]. Whereas consideration of a single bounding source term is undeniably simpler for siting and off-site planning, the results are necessarily conservative and could potentially be overly restrictive.

However, a more general interpretation, and one that is consistent with the use in international safety guides [10], is that "source term" refers to the release to the environment from any event sequence at a facility, not exclusively the bounding accident. Consideration of a broader spectrum of event sequences, and their associated source terms, allows for the safety analysis to establish an array of possible accident consequences. When combined with probabilistic insights on the event sequences, a more comprehensive picture of the overall risk of the facility becomes clearer. This is useful within more goals-setting licensing frameworks, where the regulator may have established quantitative safety objectives for different plant states, such as offsite dose limits for each of the anticipated operational occurrences, DBAs, and design extension conditions.

It is worth noting that this more general interpretation has been implicitly endorsed by the RSWG. *The Basis for the Safety Approach for Design & Assessment of Generation IV Nuclear Systems* [3], for example, states:

> "*The historical notion of a single bounding design basis accident must be replaced by a spectrum of possible accidents that represent, with a degree of confidence, the range of physical events that could conceivably challenge the plant safety*."

Furthermore, with respect to the safety approach for Gen-IV reactors, the RSWG has established [3]:

> "*Where justified, the safety approach, technical bases, and safety requirements should be goals setting and performance based to the extent practical, rather than being prescriptive*".

Most experiences determining source terms have been for postulated accidents in large, water-cooled reactors, and in many cases a large LOCA for LWRs. Through the combination of core and thermal hydraulic design, event identification/selection, and accident phenomenology, any source term associated with a Gen-IV reactor could differ from this experience in fundamental aspects like the magnitude and mix of radionuclides released, their physical and chemical forms, and the timing of the releases. There may be, therefore, significantly less-developed predictive capability as well as real data that could be used to validate models. The capability to accurately predict the source term of Gen-IV reactor accidents will be crucial in demonstrating that the safety goals of GIF have been achieved [5].

[1] This usage of DBA is potentially confusing because contemporary international safety standards like SSR-2/1 Rev. 1 [10] associate core melting (severe accidents) with design extension conditions and not design basis accidents. For this reason, this paper avoids the phrase "DBA source term" except for this historical context.

To summarize, the primary issues facing the assessment of source term for Gen-IV reactors are

1. "Source term" is not used consistently and the meaning can differ based on the context of the specific licensing regime and its history.
2. The concept of a bounding accident source term does not align with the safety approach for Gen-IV reactors.
3. Gene-IV reactors will have different source terms than most large, water-cooled reactors.

This paper therefore proposes a definition of mechanistic source term, as well as the associated methods of assessment, to address these issues.

3. Attributes of Mechanistic Source Terms

Although the concept had arguably existed for much longer, one of the earliest written definitions of mechanistic source term appears in SECY-93-092 *Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements* [11], which considered various technical and regulatory issues associated with a broad range of reactor technologies:

> "*A mechanistic source term is the result of 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 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*" [11].

The context under which this definition first appeared had two implications: (1) previously accepted source terms, like those documented in TID-14844 *Calculation of Distance Factors for Power and Test Reactor Sites* [7] for large LWRs, were "non-mechanistic," and (2) mechanistic source terms were relevant only to advanced reactors, or more specifically not to large LWRs which would continue to use "non-mechanistic" source terms like those in TID-14844 or the updated alternative NUREG-1465 *Accident Source Terms for Light-Water Nuclear Power Plants* [8]. At the least, this would be the case within the United States where this definition first appeared [11]. Setting aside this context, the original definition nevertheless succinctly describes the attributes of a mechanistic source term that would be applicable to the assessment of Gen-IV reactors. These can be separated into three fundamental components:

1. Identification of events (accident sequences) for which the source term must be evaluated.
2. Prediction of the release of radionuclides from fuel during those events.
3. Prediction of radionuclide transport, retention, and release to the environment based on phenomenological models.

The subsequent sections provide more details on each component. These descriptions borrow from the prior works of the U.S. Department of Energy including [12], [13] and [14], but this paper aims to generalize the concepts for other regulatory frameworks mainly based on IAEA safety standards.

A fundamental assumption is that, for a Gen-IV reactor for which the mechanistic source term is to be evaluated, there is a realistic representation of the entire system in hand that facilitates modelling of the phenomena at the necessary fidelity. This may include, but is not limited to, detailed design descriptions of the core region, primary heat transport system, secondary and

tertiary heat transport loops, containment, and auxiliary buildings. Only when all the systems, structures, and components (SSCs) that affect the radionuclide transport to the environment are appropriately accounted for can the source term prediction be considered mechanistic.

3.1. Radionuclide Releases from Fuel

The radionuclide inventory includes most notably fission products and their progeny. Whereas neutron activation products are likely to exist outside of the fuel, the offsite hazard from these products is usually less than that posed by the inventory of fission products. These inventories are typically determined using neutronic analysis software capable of modelling both the fission rate (power) and depletion (burnup) of the fuel elements[2]. The inventory of radionuclides within a fuel element will depend on its irradiation time and its position history within the core. This is a relatively straightforward calculation when fuel elements are in a fixed position between refueling cycles. It can become more complicated, however, when the fuel is in motion during operation, as is the case with the pebble bed very-high-temperature reactor (VHTR) or liquid-fueled molten salt reactor (MSR).

It is critical not to overlook irradiated fuel that may exist outside of the core region when determining the radionuclide inventory of the reactor. In the cases of the pebble-bed VHTR and MSR there may be circulating fuel outside of the core and in holdup volumes that must also be accounted for. Note that, by definition, “source term” refers to the releases from facilities, not just reactors, and can also include releases from spent fuel handling and storage as well as from fuel processing systems[3]. This paper reserves discussion on such releases to focus on the mechanistic source term of the Gen-IV reactors specifically.

The integrity of the fuel may be challenged in accident conditions by different thermal, mechanical, or chemical loads. “Integrity,” in this context, refers to the fuel element’s capability to contain its radionuclide inventory. The thermal, mechanical, and chemical properties of the materials that determine this integrity can also be affected by the fuel element’s irradiation history, so this may need to be considered as well. An understanding of the phenomena that affect the fuel integrity and the eventual release of the radionuclides is required to build best-estimate models. It is ultimately necessary to predict not just which radionuclides are released, but also what physical and chemical states they are in to inform the downstream transport models.

The required fidelity of the release model can depend on the specific event that is being analyzed. Mechanistic thermal-mechanical modelling of individual fuel elements with dedicated fuel analysis computer codes may be required for some DBA analysis. However, for the analysis of some beyond design basis accidents where severe fuel degradation is expected *a priori*, surrogate models of radionuclide release within integral plant response codes may be sufficient. Such surrogate release models may depend on a single parameter, like the temperature of an individual fuel assembly (group of fuel elements), and can be based on prior calculations performed with higher-fidelity fuel models.

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[2] The definition of a *fuel element* is “a rod of nuclear fuel, its cladding and any associated components necessary to form a structural entity” [6]. For coated particle fuel, such as that used in VHTR, this definition is easily extended so that a single pebble or cylindrical compact is considered a fuel element. The extension to liquid-fueled MSR is not as obvious. For the purposes of this text, a liquid fuel element may be considered as a finite volume of the solution that exists in the bulk fluid.

[3] Some MSR concepts are especially difficult to envelope within definitions that are general to all Gen-IV systems. For example, a liquid-fueled MSR with integrated reprocessing blurs the distinction between power reactor and fuel handling facility, and the accumulation of fission products separate from the fuel raises to question the greatest hazards present in the facility. Rather than attempting to answer these questions, this paper addresses only the source term of the reactor itself and leaves MST evaluation of such MSR systems for future consideration.

It is worth noting here that, for some of the Gen-IV reactors, there is not an obvious distinction between "intact" and "failed" fuel such as that indicated by the integrity of the cladding pressure boundary in LWR fuel elements. Coated particle fuel compacts in VHTR and liquid fuel in MSRs, for example, demand different definitions of fuel failure and/or degradation. In normal operation, some radionuclides may even be expected to diffuse out of, or be volatilized from, otherwise intact fuel elements, which complicates the delineation further. However, if the radionuclide release from the fuel can be accurately estimated for the postulated conditions, whether the fuel is argued to be "intact" or "failed" is immaterial to the source term prediction. Agreed definitions of "fuel damage" or "fuel degradation" should therefore not be considered prerequisites for mechanistic source term assessment.

3.2. Event Sequence Identification

The RSWG has described a systematic, risk-informed[4], technology-neutral approach for identification and categorization of event sequences to support the design and licensing of Gen-IV reactors [4]. A summary of this approach is presented here as an example of risk-informed event sequence identification, but it is acknowledged that other risk-informed approaches exist.

In the GIF approach, an event sequence is comprised of an initiating event, a sequence of plant equipment and/or operator responses, and a stable end state. The initiating event for the sequence may be internal, like an operator error or random failure of a plant SSC, or external, like an earthquake or severe weather. The sequence of responses depends on the plant design, operator actions, and any additional failure assumptions. The same initiating event could, therefore, lead to different event sequence categories.

The event sequence categories in this approach are consistent with definitions of plant states in international safety standards [10]:

- **Normal operation (NO)** – operating within specified limits and conditions, including startup, power operation, shut down, maintenance, testing, and refueling.
- **Anticipated operational occurrence (AOO)** – "A deviation of an operational process from normal operation expected to occur at least once during the operating lifetime of a facility but in which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions" [6]. The AOO category refers to event sequences resulting from the "deviation."
- **Design basis accident (DBA)** – "A postulated accident leading to accident conditions for which a facility is designed in accordance with established design criteria and conservative methodology, and for which releases of radioactive material are kept within acceptable limits" [6]. DBAs are event sequences not expected to occur in the life of a plant and are less likely than AOOs.
- **Design extension conditions (DEC)** – "Postulated accident conditions not considered for design basis accidents, but 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" [6]. DEC are very rare or complex event sequences not expected to occur over the life of a reactor fleet. They are still considered in design despite their rarity.

Risk is represented in the GIF approach as a set of frequency-consequence criteria that establish a correlation between permissible dose limits and event sequence frequency targets.

[4] The risk-informed approach combines risk insights with technical insights from traditional analyses and expert judgments. This contrasts with the risk-based approach that relies solely on risk assessments.

This frequency-consequence target is illustrated in Figure 1. Here, the risk is defined as the product of frequency and consequence, and the acceptable risk is indicated by the red "risk target" against which the risk and safety significance of the event sequences are evaluated. The specific frequency and dose limits that make up the risk target are expected to vary from country to country, but the concept is intended to be generally applicable. The core of the risk-informed approach is thus the designer demonstrating that, for the relevant design basis conditions and DEC, the system response allows the corresponding risk to be kept within the tolerable domain: below the risk-target with margin.

The risk associated with any event sequence depends on its consequence. As stated above, "consequence" in this approach is explicitly dose. Determining the offsite dose in turn requires that the release of radionuclides into the environment (source term) be known. Consistent with the *Basis for the Safety Approach for Design & Assessment of Generation IV Nuclear Systems* [3], in this approach **there is no single bounding source term but rather a spectrum of possible source terms** each with a frequency of occurrence consistent with the corresponding event sequence.

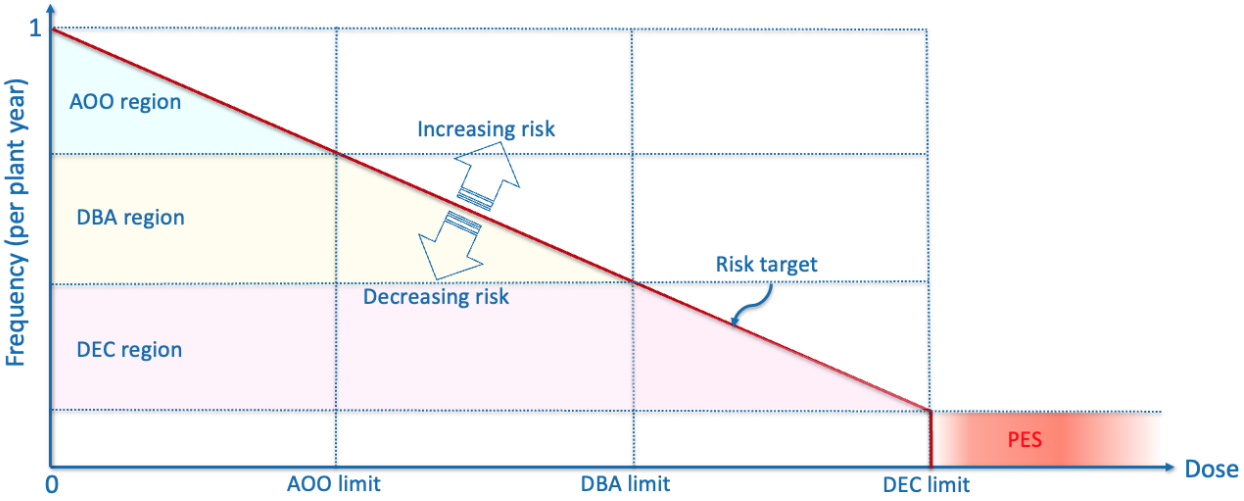


Figure 1: Concept of frequency-consequence target, reproduced from [4]. The practically eliminated situations (PES) with large or early releases are shown to be outside the plant design envelope.

Each deterministic phenomenological analysis of an event sequence that results in radionuclide releases to the environment is, in principle, a mechanistic source term estimation. Considering that a mechanistic source term is created using "best-estimate" models by definition, this may raise the question of how the required conservatism of design basis analysis is achieved. It is accepted that use of "best-estimate" models in deterministic safety analysis does not necessarily preclude the analysis result from being conservative depending on the assumptions of the initial and boundary conditions and/or the treatment of uncertainties[5] [15]. Conservatism in the risk-informed approach largely comes from the imposed restrictions on the responding plant equipment, based on its classification, which is used to define the bounding event sequences above. For example, for analysis of DBA, only the classified safety systems are assumed to be available.

[5] Table 1 of Specific Safety Guide No. SSG-2 (Rev. 1) Deterministic Safety Analysis for Nuclear Power Plants [15] presents different options. The "best estimate plus uncertainty" (BEPU) method, for example, has been applied to conservative deterministic safety analysis.

3.3. Radionuclide Transport, Retention, and Release to the Environment

The transport of radionuclides from fuel to the environment in an event sequence is determined by phenomena affecting not only the transport of fission products but also the integrity of any barriers. The release from the fuel represents the failure of merely one physical barrier of many that are part of the design according to the principle of defense in depth. It is difficult to comprehensively describe all the phenomena that may affect radionuclide transport because of the fundamental differences between Gen-IV reactor technologies. The various systems and their behaviors that need to be captured in an event sequence may include:

- Thermal hydraulic response of the primary heat transport system, reactor cavity, containment and/or confinement buildings.
- Fuel heat up, chemical reactions, fuel degradation, and relocation.
- Heat up of the reactor vessel, thermal and mechanical loading, failure of the vessel, and transfer of core materials to the reactor cavity.
- Reactions between core material and concrete.
- In-vessel and ex-vessel production, transport, and reaction of combustible materials.
- Chemical reactions affecting the mobility, volatility, and retention of fission products, particularly iodine.
- Fission product aerosol and vapor transport, deposition, and resuspension or revolatilization.
- Impacts of classified safety systems and/or safety features on any of the above.

Capturing these behaviors requires phenomenological models of heat and mass transfer, thermomechanics, and thermochemistry, just to name a few. The computer codes that model these phenomena should predict the transient evolution of an event sequence, including determining how and when certain release barriers are compromised, and the subsequent transport of radionuclides through one or more of the compromised barriers up to their release to the environment.

Accident conditions at a nuclear power plant are characterized by many interacting processes and phenomena. The toolset used for predicting mechanistic source term may, therefore, be comprised of several loosely or tightly coupled computer codes that each focus on a specific technical discipline, like thermal hydraulics, fission product release from fuel, and fission product aerosol and/or vapor transport in containment. Experience has also shown that, at least for severe plant conditions, the dominant uncertainties affecting the accident progression and source term predictions are in the interaction of processes and phenomena at the reactor scale [14]. This has driven the development of integral plant response codes that are intended to characterize the full system response in a best-estimate manner to support risk-informed decision making.

Although the topic was not explicitly addressed in the GIF risk-informed approach paper [4], this paper asserts that mechanistic source term assessment is intrinsic to the GIF approach to safety where dose is the specific consequence or relevance. The evaluation of mechanistic source terms, including the retention of radionuclides by barriers and the transport of radionuclides through barriers and pathways to the environment, has already been described as a required element of an integrated, risk-informed, performance-based approach to designing and licensing non-LWRs in the United States [16].

4. Technical Considerations for the Assessment of Mechanistic Source Terms

Thus far, this paper has presented the historical context for source term assessment, as well as the different components of mechanistic source term predictions for Gen-IV reactors. This section briefly outlines the basic considerations that must be made in a mechanistic source term assessment for it to be consistent with the GIF risk-informed approach [4] and, more broadly, the approach for the safety design and assessment of Gen-IV systems [3].

A mechanistic source term for a Gen-IV reactor is the result of an analysis of radionuclide release based on the challenges to physical barriers from the specific event sequence being evaluated. It is developed using best-estimate phenomenological models of the transport of fission products from the fuel, through all holdup volumes and boundaries, up to release into the environment. The following conditions should apply when developing a mechanistic source term:

1. The performance of the reactor fuel in operational states and accident conditions must be sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist from research and development and testing programs to provide adequate confidence in the mechanistic approach.
2. The transport of fission products can be adequately modelled for all barriers and pathways to the environment. The physical models should be sufficiently realistic so that the values and limitations of any mechanism or barrier are not obscured.
3. The event sequence in the analysis is identified in consideration of both deterministic expert judgement and risk insights. Assumptions on the initial and boundary conditions, the bounding of uncertainties, and the responding plant equipment, are made consistently with the analysis requirements for the event sequence category.
4. The resulting source term includes the timing, physical and chemical form, and magnitude of radionuclide releases to the environment for a specific event sequence so that the offsite consequence of the event sequence can be determined.

The mechanistic source term prediction should be justified by validating models and by accounting for uncertainties, including uncertainties in both the plant parameters as well as in the physical models. Generally speaking, the qualification of computer codes for nuclear safety analysis has included verification, validation, and uncertainty quantification. Therefore, to successfully implement the risk-informed approach to safety, **qualified analytical tools will be required for the assessment of Gen-IV reactor mechanistic source terms**. “Qualified” in this context means demonstrated to be capable of predicting source terms with the accuracy and fidelity required by the objectives of the analysis. Where such analytical tools are not already available, the qualification of source term computer codes is likely to involve gap assessments and subsequent analysis and testing programs to address any gaps as required to meet regulatory requirements. The size of any gaps, and the magnitude of an analysis and testing effort to address them, will depend strongly on the particular Gen-IV reactor technology and specific plant design. However, considering the already limited experience predicting source terms for Gen-IV reactors compared to the previous-generation LWRs, both the tools and their validation bases are likely to require development.

5. Conclusions for Mechanistic Source Term Assessment of Generation-IV Reactors

The RSWG has investigated mechanistic source term assessment methodologies for Gen-IV energy systems. A preliminary review determined that (1) “source term” is not used consistently and the meaning can differ based on the context of the specific licensing regime and its history, (2) the concept of a bounding accident source term does not align with the safety approach for

Gen-IV reactors, and (3) Gen-IV reactors will have different source terms than most large, water-cooled reactors. Definition of “mechanistic source term” was sought to address these issues. A mechanistic source term for a Gen-IV reactor is the result of an analysis of radionuclide release based on the challenges to physical barriers from the specific event sequence being evaluated. It is developed using best-estimate phenomenological models of the transport of fission products from the fuel, through all holdup volumes and boundaries, up to release into the environment.

This paper described the attributes of mechanistic source terms, which included identification of events (accident sequences) for which the source term must be evaluated, prediction of the release of radionuclides from fuel during those events, and prediction of radionuclide transport, retention, and release to the environment based on best-estimate phenomenological models. Technical considerations for the assessment of mechanistic source terms for Gen-IV reactors were also proposed.

In summary, this assessment concluded the following:

1. There is no single bounding source term but rather a spectrum of possible source terms.
2. Mechanistic source term assessment is intrinsic to the risk-informed approach to safety previously described by the GIF RSWG.
3. Qualified analytical tools will be required for the assessment of Gen-IV reactor mechanistic source terms.

The final conclusion above may represent an actionable item for GIF. The tools for predicting mechanistic source term in each Gen-IV system could be more thoroughly investigated to determine the capabilities of models and their qualification bases. Gap assessments could be performed to determine what, if any, new analysis and testing programs may be required so that mechanistic source term predictions could be accepted in risk-informed safety and licensing assessments. Such investigations on specific Gen-IV systems are outside the mandate of the RSWG but could be executed by projects under the current GIF system arrangements or by a new dedicated task force.

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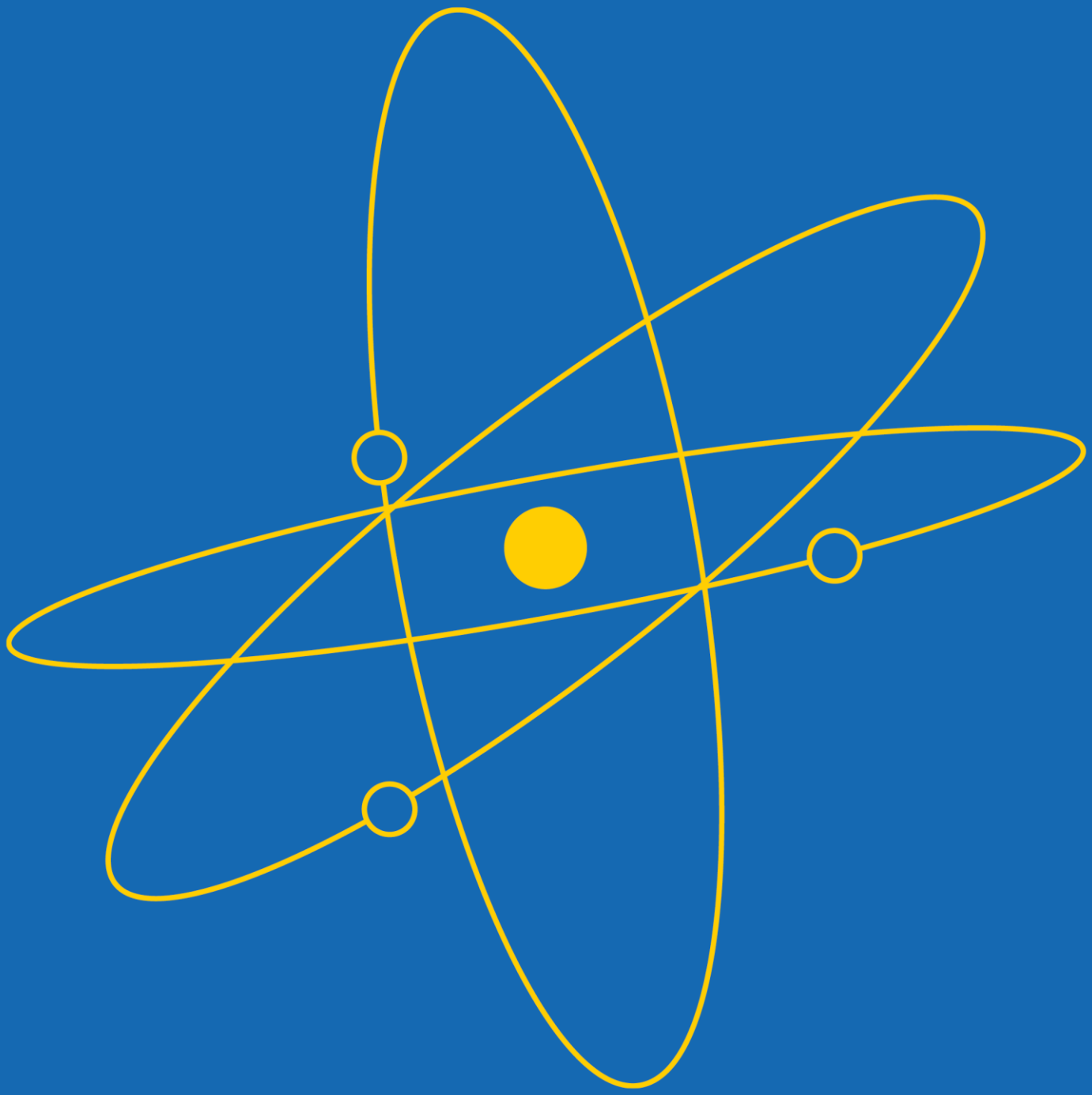
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THE GENERATION IV INTERNATIONAL FORUM

Established in 2001, the Generation IV International Forum (GIF) was created as a co-operative international endeavor seeking to develop the research necessary to test the feasibility and performance of fourth generation nuclear systems, and to make them available for industrial deployment by the 2030s. Under the new 2025 GIF Framework Agreement, GIF brings together countries, as well as Euratom, representing 27 EU member states, to co-ordinate research and develop these systems. GIF has selected six reactor technologies for further research and development: the gas-cooled fast reactor (GFR), the lead-cooled fast reactor (LFR), the molten salt reactor (MSR), the sodium-cooled fast reactor (SFR), the supercritical-water-cooled reactor (SCWR) and the very-high-temperature reactor (VHTR).



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