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A Methodology for the Integration of a Mechanistic Source Term Analysis in a Probabilistic Framework for Advanced Reactors

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A METHODOLOGY FOR THE INTEGRATION OF A MECHANISTIC SOURCE TERM ANALYSIS IN A PROBABILISTIC FRAMEWORK FOR ADVANCED REACTORS

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ABSTRACT

GE Hitachi Nuclear Energy (GEH) and Argonne National Laboratory are currently engaged in a joint effort to modernize and develop probabilistic risk assessment (PRA) techniques for advanced non-light water reactors. At a high level, the primary outcome of this project will be the development of next-generation PRA methodologies that will enable risk-informed prioritization of safety- and reliability-focused research and development, while also identifying gaps that may be resolved through additional research. A subset of this effort is the development of PRA methodologies to conduct a mechanistic source term (MST) analysis for event sequences that could result in the release of radionuclides. The MST analysis seeks to realistically model and assess the transport, retention, and release of radionuclides from the reactor to the environment. The MST methods developed during this project seek to satisfy the requirements of the Mechanistic Source Term element of the ASME/ANS Non-LWR PRA standard. The MST methodology consists of separate analysis approaches for risk-significant and non-risk significant event sequences that may result in the release of radionuclides from the reactor. For risk-significant event sequences, the methodology focuses on a detailed assessment, using mechanistic models, of radionuclide release from the fuel, transport through and release from the primary system, transport in the containment, and finally release to the environment. The analysis approach for non-risk significant event sequences examines the possibility of large radionuclide releases due to events such as re-criticality or the complete loss of radionuclide barriers. This paper provides details on the MST methodology, including the interface between the MST analysis and other elements of the PRA, and provides a simplified example MST calculation for a sodium fast reactor.

INTRODUCTION

A joint effort between General Electric–Hitachi Nuclear Energy (GEH) and Argonne National Laboratory is currently underway which intends to develop and modernize probabilistic risk assessment (PRA) techniques for advanced non-light water reactors (LWRs). The primary outcome of this project will be the development of a next-generation PRA that will satisfy anticipated regulatory requirements and enable risk-informed prioritization of safety- and reliability-focused research and development, while also identifying gaps that may be resolved through additional research. Previous efforts to construct a PRA for GEH's Power Reactor Inherently Safe Module (PRISM) (ref. 1), a sodium-cooled fast reactor (SFR), are being leveraged as a foundation for this project. Additionally, this effort is being executed in accordance with guidance provided by the recently issued ASME/ANS Non-LWR PRA (2) standard, which has been approved for trial use.

A subset of this effort is the development of PRA methodologies to conduct a mechanistic source term (MST) analysis for event sequences that could result in the release of radionuclides. The MST analysis seeks to realistically model and assess the transport, retention, and release of radionuclides from the reactor to the environment. This report provides details on the developed MST methodology and how the methodology satisfies the requirements of the ASME/ANS PRA standard for advanced Non-LWRs.

The first section of this work describes the requirements of the new ASME/ANS Non-LWR PRA standard, which contains a mechanistic source term (MS[1]) element; this standard element is directly applicable to the current analysis. The guidance of the Non-LWR PRA standard is vital for this effort, as no operating U.S. reactors have performed an MST analysis,

[1] In the current work, MST is used as the acronym for mechanistic source term, and MS is used to refer to the mechanistic source term element of the Non-LWR PRA standard.

and regulatory requirements of such an analysis have not yet been clarified.

The following section provides details regarding the MST methodology, and discusses how the methodology fulfills the requirements of the Non-LWR PRA standard. The methodology provides a pathway for determining the event sequences that require detailed analyses, which focus heavily on computational modeling of core damage.

The penultimate section provides an application of the MST methodology using an example from the PRISM PRA. In particular, core conditions, and the possibility of fuel damage resulting from a given accident sequence, are predicted using the SFR severe accident code SAS4A/SASSYS-1 (3). Assessing the transport of radionuclides requires the integration of a variety of approaches, as no single computational tool currently exists to complete this task.

ASME/ANS PRA STANDARD FOR NON-LWRS

A key focus of the current non-LWR PRA modernization and development effort is the establishment of analysis techniques that satisfy the requirements of the recently completed ASME/ANS PRA Standard for Advanced Non-LWR Nuclear Power Plants. The Non-LWR PRA standard, which was approved for a 36-month trial use period in 2013, is the only such PRA standard document specifically created for advanced non-LWRs. Unlike the ASME/ANS PRA standards for LWRs, which address only a portion of the total PRA per each standard document, the Non-LWR PRA standard is a comprehensive document that includes all plant initiators and hazards, and covers the analysis from initiating event to offsite consequence analysis.

As with the ASME/ANS standards for LWRs, the Non-LWR PRA standard is divided into PRA elements, which are composed of high level requirements (HLRs) and supporting requirements (SRs). If each SR is fulfilled, then the HLR is fulfilled. If all HLRs are met, then the PRA element is satisfied. In total, the Non-LWR PRA standard contains 18 PRA elements, with over 200 HLRs and more than 1000 SRs for all elements.

The Non-LWR PRA standard is also designed to address PRAs during the various stages of reactor design and development. Many SRs are split into multiple Capability Categories (CCs), which vary depending on the degree of realism and plant-specificity needed for the analysis. The proper CC is selected through the PRA application process, where factors such as the life-cycle stage, site characteristics, and PRA application are assessed to determine the correct CC. For the current project, the goal is to satisfy CC II of the pertinent SRs, which are described below:

- **Scope and level of detail:** Resolution and specificity sufficient to identify the relative importance of the contributors at the component level and associated human actions, as necessary, and relevant physical phenomena and release characteristics.
- **Plant-specificity:** Use of plant-specific data/models for the significant models.
- **Realism:** Departures from realism will have small impact on the conclusion and risk insights supported by good practices.

Mechanistic Source Term Requirements

The Non-LWR PRA standard contains a separate PRA element regarding the mechanistic source term (MS) assessment, which contains 5 HLRs and 26 SRs[2]. All HLRs and SRs are considered applicable to the current work. For complete requirement details, see Section 4.5.16 of ref. (2).

HLR-MS-A describes how the MS release categories must be consistent with the event sequence analysis, and defines the characteristics that must be included when specifying the release categories.

HLR-MS-B focuses on the process for assessing the source term for each release category, including selecting the appropriate radionuclide transport models and selecting proper inputs.

HLR-MS-C describes the process for quantifying the MS assessment, which includes a description of the phenomena and plant systems that must be considered, as well as the necessary details to characterize a radionuclide release[3].

HLR-MS-D focuses on the MS uncertainty evaluation, including the proper treatment of alternative sources of uncertainty (parameter and model), and types of uncertainty (aleatory and epistemic).

Lastly, HLR-MS-E sets the documentation requirements for the MS analysis, including the reporting of assumptions, technical basis, and limitations. This HLR is included for all PRA elements, and is a necessary step for a complete PRA.

METHODOLOGY

This section provides a high-level review of the methodology to be used in the MST evaluation. Table 1 provides an overview of the process, which begins with a series of steps to determine the type of MST assessment necessary for the event sequence, followed by a bifurcation between risk significant sequences (RSSs) and non-risk significant

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[2] The Non-LWR PRA standard currently only considers the reactor core as a source of radionuclides for the MST assessment. Although the current PRISM PRA update and modernization is not analyzing releases from auxiliary systems, possible radionuclide releases from the cover gas or primary sodium, even in the event of no core damage, are considered for this effort. The guidance provided from the Non-LWR PRA standard for the assessment of radionuclide releases from the core is used for these ex-core release scenarios, even though they are outside the intended scope of the standard.

[3] The Non-LWR PRA standard makes several references to consideration of the reactor coolant pressure boundary (RCPB). However, for a pool-type SFR, there is no RCPB, as the primary system is at atmospheric pressure. Instead, the term reactor coolant boundary (RCB) or primary reactor coolant boundary (PRCB) is often used to specify the boundary of the primary system, as well as to clarify that the pressure difference across the boundary is low.

Table 1. Mechanistic Source Term Evaluation Procedure

Step	Description	
1	Identify event sequences	
2	Preliminary event sequence assessment	
3	Preliminary determination of event sequence outcome (possible radionuclide release) and risk significance	
	RSSs with Possible Radionuclide Release	***Non-RSSs with Possible Radionuclide Release***
4	Determination of radionuclide inventory	Determination of radionuclide inventory
5	Mechanistic modeling of core response	Simplified modeling of core response
6	Mechanistic modeling of plant response	Simplified modeling of plant response
7	Binning of similar radionuclide release sequences	Binning of similar radionuclide release sequences
8	Mechanistic modeling of offsite release	Simplified modeling of offsite release
9	Reassessment of event sequence risk significance	

sequences (non-RSSs). The remainder of this section details each step of the MST evaluation procedure, and reviews how the requirements of the Non-LWR PRA standard are satisfied.

- ***Step 1****:* The event sequences to be analyzed are identified as part of the PRA development.

- ***Step 2****:* A preliminary event sequence assessment is made. This step may involve a simplified plant model, or engineering judgment. The purpose of the initial event sequence assessment is to determine if the sequence may result in radionuclide release from the plant (and hence require a more detailed source term assessment), and whether the scenario may be risk significant.

- ***Step 3****:* Utilizing the results of the preliminary event sequence assessment, the sequences are separated into different classifications. A possible classification scheme is the following:

  - RSS with possible radionuclide release[4]
  - Non-RSS (low frequency event sequence) with possible radionuclide release
  - Non-RSS (low frequency event sequence) with no possibility of radionuclide release

  For those sequences that are in the first two categories (possible radionuclide release), additional source term calculations are conducted, and the procedure bifurcates. No source term analysis is conducted for non-RSSs with no possibility of radionuclide release.

- ***Step 4****:* For both categories of events with possible radionuclide release (RSS and non-RSS) the radionuclide inventory at the start of the event sequence is determined for use in subsequent radionuclide release and transport calculations. For RSSs, this may involve a more-detailed analysis than non-RSSs, as a thorough mechanistic assessment of the event sequence will follow.

- ***Step 5****:*
  RSS with the Possibility of Radionuclide Release
  The response of the core to the event sequence is mechanistically modeled using a best-estimate computer code (or similar). The goal of the analysis is to determine the condition of the fuel within the core and to characterize any released radionuclides.

  Non-RSS with the Possibility of Radionuclide Release
  A simplified event sequence analysis is conducted, which may or may not utilize mechanistic models. For example, in an SFR, a non-RSS may result in possible recriticality and energetics. This may warrant a degree of mechanistic modeling, even though the event has a very low frequency or is not risk significant.

- ***Step 6****:*
  RSS with the Possibility of Radionuclide Release
  The response of the plant system to the event sequence, including the possible degradation of radionuclide barriers is mechanistically modeled. The focus of this step is the transport and retention of radionuclides as they move through the reactor system and interact with barriers. The output of the analysis is the inventory and characteristics of radionuclides that are available for offsite dispersion.

  Non-RSS with the Possibility of Radionuclide Release
  A simplified plant response analysis is conducted for the event sequence. The focus of this analysis is on the potential magnitude of radionuclide release. Even if an event sequence is non-RSS, a large potential radionuclide release may result in the need for additional analyses.

- ***Step 7****:* For both categories of events with possible radionuclide release (RSSs and non-RSSs) binning of

[4] The determination of risk significance is project dependent, but should be consistent with the definition in Section 2.2 of the Non-LWR PRA standard (2). The preliminary assessment of risk significance may be based only on sequence frequency, then revised during step 9 to include offsite consequence. Also, it is assumed that all RSSs involve the possibility of radionuclide release, as the offsite risk would be essentially zero for event sequences without possible release.

event sequences with similar radionuclide release characteristics is conducted[5]. The output of this step is a reduced number of release categories to be modeled as part of the offsite dispersion analysis.

- ***Step 8***:
  RSS with the Possibility of Radionuclide Release
  A mechanistic offsite dispersion analysis is conducted, likely using a detailed computational tool, for the resulting release categories of Step 7.

  Non-RSS with the Possibility of Radionuclide Release
  A simplified offsite dispersion analysis may be conducted to gauge the impact of a non-risk significant event, such as determining the consequences of an extremely low-frequency energetic event.

- ***Step 9***: Based on the findings of steps 4 through 8, the risk significance of event sequences is reassessed. It is possible that sequences that were originally placed in the "non-RSS with possible radionuclide release" category are found to be risk significant as a result of the simplified source term assessment (steps 4 through 8). If this is the case, the sequence will then be reevaluated using the "RSS with possible radionuclide release" procedure, where detailed mechanistic modeling is performed.

Throughout steps 4 through 8, uncertainties (both model and parameter) are identified. For the mechanistic calculations as part of the RSS procedure, uncertainty analyses are also performed to gauge their impact on the resulting release. Simplified uncertainty assessments, such as sensitivity studies and scoping calculations, may be performed for non-RSSs with radionuclide release.

Non-LWR PRA Requirements

Tables 2 through 6 describe how the proposed MST methodology satisfies the requirements of the ASME/ANS PRA standard for advanced Non-LWRs.[6]

Table 2. Process to Satisfy Non-LWR PRA Standard Requirement MS-A.

Index No.	Process to Satisfy Requirement
MS-A1	Steps 1 through 3 of the procedure outlined in Table 1 identify the event sequences that require consideration for the MST analysis.
MS-A2	Steps 4 through 6 of the procedure outlined in Table 1 model the event sequence and provide the necessary characteristics of the radionuclide release.
MS-A3	See MS-A2 (also, the Application section of the current report provides additional detail on the radionuclide release characteristics detailed as part of the analysis).
MS-A4	Binning will only be performed if the radionuclide releases from separate event sequences are similar in nature, as to have identical (or nearly identical) offsite dispersion characteristics.

Table 3. Process to Satisfy Non-LWR PRA Standard Requirement MS-B.

Index No.	Process to Satisfy Requirement
MS-B1	Proper modeling techniques will be utilized (see the Application section of the current report for modeling details).
MS-B2	MS model inputs will be plant-specific.
MS-B3	Step 3 of the procedure outlined in Table 1 determines the risk significance of event sequences, but if binning of event sequences is performed, variation of release characteristics within a bin will be examined to assure that the representative release is properly defined for the bin.
MS-B4	Best available model/data/computer codes will be utilized for the MST analysis (see the Application section of the current report for modeling details).

Table 4. Process to Satisfy Non-LWR PRA Standard Requirement MS-C.

Index No.	Process to Satisfy Requirement
MS-C1	Step 4 of the procedure outlined in Table 1 quantifies and characterizes the plant-specific radionuclide inventories.
MS-C2	Barriers to release will be identified throughout steps 5 and 6 of Table 1.
MS-C3	Transport phenomena and characteristics will be identified (see the Application section of the current report for transport details).
MS-C4	Applicable phenomena for metal fuel, pool type sodium fast reactors will be included in the MST analysis.
MS-C5	The bifurcation that occurs after step 3 of Table 1 determines the proper level of MST detail depending on the risk significance of the event sequence.
MS-C6	Plant-specific MST analyses will be performed (See the Application section of the current report describes the plant-specific models).
MS-C7	Best-estimate (realistic) radionuclide release evaluations will be performed.
MS-C8	Radionuclide release characteristics, including those listed in the standard, will be defined when applicable.

[5] Binning of event sequences may also be possible after Step 5, if similar event sequences are found.

[6] For requirement details, see Section 4.5.16 of ref. (2).

Table 5. Process to Satisfy Non-LWR PRA Standard Requirement MS-D,

Index No.	Process to Satisfy Requirement
MS-D1	Parameter uncertainties affecting the MS evaluation will be identified and addressed.
MS-D2	Key sources of model uncertainty will be identified and addressed.
MS-D3	Uncertainty analyses will be performed to gauge the impact of the identified uncertainties.
MS-D4	Key uncertainty sources will be characterized with a probability distribution.
MS-D5	The evaluation of epistemic and aleatory uncertainties will be performed throughout the analysis procedure in a fashion that allows for the evaluation of their impacts separately.
MS-D6	Sensitivity analyses will be performed for those uncertainties that cannot be properly characterized with a probability distribution.

Table 6. Process to Satisfy Non-LWR PRA Standard Requirement MS-E,

Index No.	Process to Satisfy Requirement
MS-E1	All documentation procedures will be maintained in accordance with the requirements of the standard and GE-Hitachi protocol.
MS-E2	
MS-E3	The results of the quantitative uncertainty evaluation will be provided.
MS-E4	All documentation procedures will be maintained in accordance with the requirements of the standard and GE-Hitachi protocol.

APPLICATION

The following section describes the application of the MST methodology reviewed in the preceding section for the development and modernization of the PRISM PRA. Graphically, the process follows the procedure outlined in Figure 1. The remainder of this section describes each step of the process, and includes a simplified example.

Step 1: Event Sequence Identification

In the first step, the event sequences to be evaluated are identified, as part of PRA event tree (event sequence) development process.

Step 2: Preliminary Event Sequence Assessment

A preliminary assessment of the event sequences is performed. For the PRISM PRA update and modernization, this task is performed using expert judgment and/or a limited number of SAS4A/SASSYS-1 code simulations. The goal is to determine whether radionuclide release is possible for the event sequence under consideration, and to establish the preliminary risk significance of the event sequence[7].

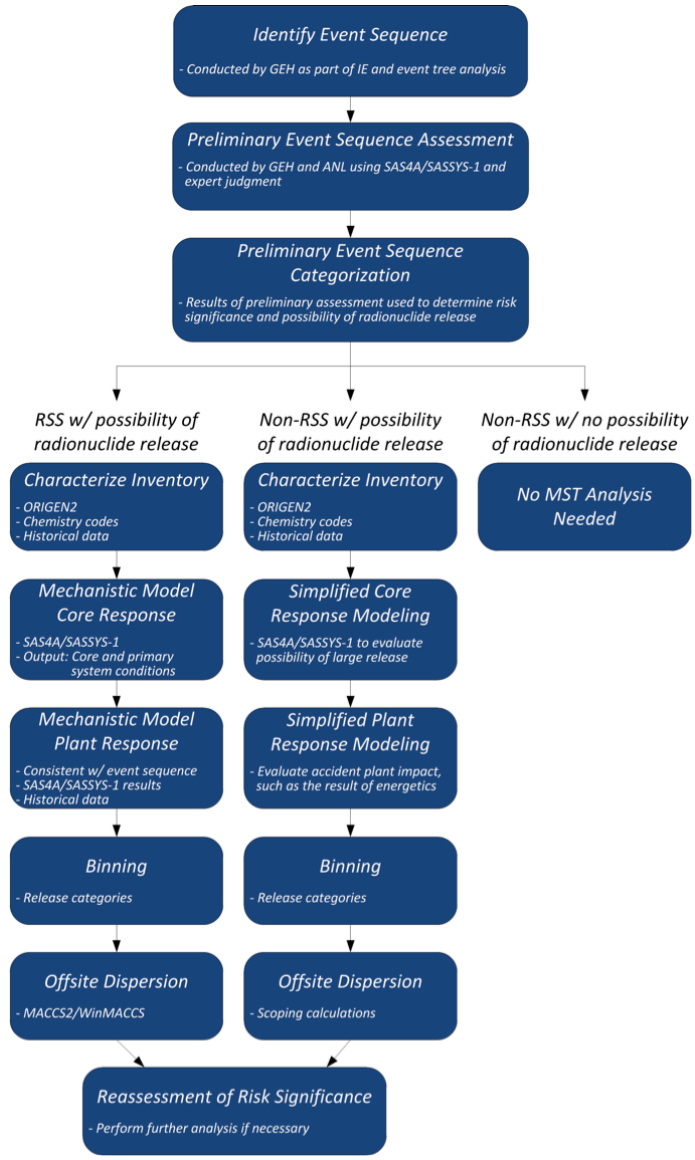

Figure 1. PRISM PRA MST Procedure

The conclusions may change as additional analyses are conducted, but this step is necessary for the grouping of event sequences in Step 3. The preliminary event sequence

[7] The risk significance of an event sequence can be determined in various ways. The Non-LWR PRA standard defines a significant event sequence as:

One of the set of event sequences included in a PRA model, defined at the functional or systematic level, that, when rank-ordered by decreasing frequency, contributes a specified percentage of the RCF (release category frequency), or that individually contributes more than a specified percentage of the RCF or other risk metric calculated in the PRA. Depending on the context, significance may be measured in terms of the total integrated risk or for the risk associated with a specific source of radioactive material, POS, and hazard group. For this version of the standard, the aggregate percentage for the set is 95%, and the individual event sequence percentage is 1%. Event sequence significance can be measured relative to each separate RCF. For hazard groups that are analyzed using methods and assumptions that can be demonstrated to be conservative or bounding, alternative numerical criteria may be more appropriate and, if used, should be justified.

assessment also coincides with the development of *simplified* success criteria, as described in ref (3).

Step 3: Preliminary Event Sequence Categorization

The results of the preliminary event sequence assessment are used to group the event sequences into a series of categories, which determine the subsequent source term analysis method. For the PRISM PRA update and modernization, the initial event sequence categories are as follows:

1) RSS with Possibility of Radionuclide Release
2) Non-RSS with Possibility of Radionuclide Release
3) Non-RSS with No Possibility of Radionuclide Release

The event sequences in the first two categories will undergo additional source term calculations, with different MST treatments between the two. This step involves the results of the preliminary event sequence modeling and the preliminary event tree quantification.

Step 4: Inventory Characterization

For both RSSs and non-RSSs with possible radionuclide release, the radionuclide inventory is determined. For the PRISM PRA update and modernization, this process begins with the use of ORIGEN2 (5) to determine the radionuclide isotopes present in the fuel at the time of the accident (which is a function of burnup, fuel location, time-in-cycle, etc.)[8]. The quantity of individual elements and isotopes present in the fuel is important in determination of the magnitude of radionuclide release.

The ORIGEN2 calculation may be supplemented with chemical thermodynamic equilibrium code calculations, using programs such as HSC Chemistry (6) or FactSage (7), and the findings from past metal fuel experience, to determine the chemical form of the elements in the fuel. The chemical form is important, as it can influence radionuclide retention and release during fuel failure events.

Step 5: Core Response Modeling

RSS with Possibility of Radionuclide Release

Mechanistic modeling of the event sequence is conducted to determine the core response to the accident sequence. For the PRISM PRA update and modernization, this task is completed using SAS4A/SASSYS-1, which is capable of modeling conditions in the core and predicting fuel failure.

SAS4A/SASSYS-1 contains several modules that are capable of treating the various phases of fuel pin failure. The phenomena treated by SAS4A/SASSYS-1 include: clad stress and failure induced by eutectic penetration and mechanical strain; fuel and clad mechanical deformation due to thermal influences and fission gas generation; in-pin molten fuel relocation; and ex-pin fuel movement. The severe accident tool is also capable of modeling sodium boiling and the associated neutronic and thermal hydraulic effects. The majority of severe accident models implemented in SAS4A/SASSYS-1 have been validated for certain scenarios through comparisons with tests performed at EBR-II and in the M-Series tests at TREAT (8).

The results of each SAS4A/SASSYS-1 simulation provide a detailed description of the condition in each core channel, including temperatures at axial and radial locations in the fuel and clad, along with axial temperature distributions in the coolant for each channel. This information can be used to establish ranges of possible fuel damage conditions.

SAS4A/SASSYS-1 also reports a variety of additional information which can be utilized in the MST calculation, if need be. For core conditions, the code will also output an axial distribution of the margin to sodium boiling and margin to cladding failure for each fuel channel. For plant conditions, SAS4A/SASSYS-1 provides primary sodium temperatures and pressures, reactor vessel and guard vessel temperatures, and cover gas temperature and pressure.

Throughout step 5, uncertainties will be addressed according to the requirements of the Non-LWR PRA standard. Parameter uncertainties will be characterized to the extent possible and directly propagated through the use of multiple SAS4A/SASSYS-1 simulations. For model uncertainty or parameter uncertainties that cannot be fully characterized, sensitivity analyses will be performed to gauge their impact on the radionuclide release results.

Non-RSS with Possibility of Radionuclide Release

Simplified core response modeling will be conducted for the non-RSSs with the possibility of radionuclide release. The main objective of this analysis is to determine if any of the non-RSSs may lead to a large radionuclide release, such as those that may result from an energetic event. It also allows for the identification of any possible "cliff-edge" effects, where a slight variation in the event sequence or event parameter leads to a significant increase in the offsite consequence.

For most non-RSSs, the simplified core response modeling for the PRISM PRA update and modernization involve a limited number of SAS4A/SASSYS-1 calculations. The calculations explore the limits of the uncertainty space to determine if a large core damage event is possible and whether additional analyses may be necessary to examine recriticality or energetics

Step 6: Plant Response Modeling

RSS with Possibility of Radionuclide Release

Mechanistic modeling of the event sequence is conducted to determine the plant response to the accident scenario. For the PRISM PRA update and modernization, this step requires a combination of modeling tools and approaches, since no single computational tool currently exists to model the transport of

[8] This applies only to event sequences where the possible radionuclide release is from core damage. Other source characterization techniques may be needed for radionuclide releases from other sources, such as the primary sodium or cover gas region.

radionuclides from through the reactor system and plant for an SFR[9].

First, the results of the core response modeling in step 5 are reviewed to determine the extent of possible radionuclide release from the fuel, and to establish factors that could influence radionuclide transport. SAS4A/SASSYS-1 also provides other information about the state of the reactor system, which should be considered to assess the impact on safety. For example, if SAS4A/SASSYS-1 modeling results show that the reactor vessel is at extremely high temperatures for a prolonged period due to a loss of decay heat removal, additional analysis examining vessel integrity may be needed. This event would provide additional leakage pathways for radionuclides to escape the primary sodium system in the event of core damage.

Second, the event sequence described in the event tree is reviewed to check for any postulated events that could affect the transport of radionuclides. For example, if the initiating event for the sequence is a large earthquake, the analysis should consider the possibility of reactor head or containment damage, which could impact the retention and transport of radionuclides.

Third, past experimental and accident data (such as those reviewed as part of ref (9) are used to determine approximate release fractions of radionuclides from the fuel to the primary sodium, from the sodium to the cover gas, and from the cover gas into containment. As a mechanistic computer model does not currently exist to complete this task, historical data offer the best available path forward[10].

Non-RSS with Possibility of Radionuclide Release

The goal of the plant response for non-RSSs is again to estimate the likelihood of a large radionuclide release or identify cliff-edge effects. For example, if energetic sequences are found to occur, simplified analyses of reactor vessel and containment integrity may be warranted. As with step 5, the bounds of the uncertainty space will be explored to investigate the range of possible results.

Step 7: Event Sequence Binning

For both RSSs and non-RSSs with the possibility of radionuclide release, binning of the event sequences occurs to reduce the number of offsite dispersion analyses necessary. The event sequence grouping will only occur if there is adequate similarity in the radionuclide releases of sequences (in terms of radionuclides, timing, release pathway, etc.). It should also be noted that binning of event sequences may also occur following the completion of step 5, if the similarity in core and plant damage states allows it.

Step 8: Offsite Dispersion

RSS with Possibility of Radionuclide Release

Using the radionuclide inventories determined for each release category, offsite dispersion can be modeled with the WINMACCS/MACCS code (10). Uncertainties affecting the offsite dispersion, such as meteorological conditions, will be considered as necessary.

Non-RSS with Possibility of Radionuclide Release

For non-RSSs, offsite dispersion calculations may be conducted to explore the range of possible releases and their consequences. This may be necessary if some non-RSSs appear to be close to risk significance upon more detailed examination.

Step 9: Reassessment of Risk Significance

The risk significance of the event sequences analyzed as part of the MST procedure is reassessed to ensure that event sequences have received the proper MST analysis treatment. For example, an event sequence may originally be categorized as a non-RSS in step 3, due to a very low frequency of occurrence. However, during steps 4 through 8, it may be found that a very large radionuclide release is possible with that event sequence. Therefore, the event sequence should be considered a RSS, even though the frequency is low. If this is the case, then the event sequence should be re-analyzed using the "RSS with Possibility of Radionuclide Release" procedure for steps 4 through 8.

SUMMARY AND CONCLUDING REMARKS

As part of an effort to modernize and develop PRA techniques for advanced non-LWRs, a mechanistic source term analysis methodology was developed, which seeks to realistically analyze the transport and release of radionuclides. The MST methodology analyzes each event sequence of the PRA to determine the possibility of radionuclide release and the risk significance of the sequence. These factors are used to determine the level of detail of necessary for the radionuclide release and transport calculation. The methodology described throughout this paper satisfies the mechanistic source term (MS) element of the recently issued ASME/ANS Non-LWR PRA Standard.

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[9] Although an integral SFR plant analysis radionuclide transport code does not currently exist, there are efforts underway at Sandia National Laboratories to migrate the modeling capabilities of CONTAIN-LMR into MELCOR. However, for the project discussed here, the new version of MELCOR was not yet available.

[10] See ANL-ART-38 (11).

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