Advanced Reactor PSA Methodologies for System Reliability Analysis and Source Term Assessment

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Abstract. Beginning in 2015, a project was initiated to update and modernize the probabilistic safety assessment (PSA) of the GE-Hitachi PRISM sodium fast reactor. This project is a collaboration between GE-Hitachi and Argonne National Laboratory (Argonne), and funded in part by the U.S. Department of Energy. Specifically, the role of Argonne is to assess the reliability of passive safety systems, complete a mechanistic source term calculation, and provide component reliability estimates. The assessment of passive system reliability focused on the performance of the Reactor Vessel Auxiliary Cooling System (RVACS) and the inherent reactivity feedback mechanisms of the metal fuel core. The mechanistic source term assessment attempted to provide a sequence-specific source term evaluation to quantify offsite consequences. Lastly, the reliability assessment focused on components specific to the sodium fast reactor, including electromagnetic pumps, intermediate heat exchangers, the steam generator, and sodium valves and piping.

Key Words: PSA, Sodium Fast Reactor, Passive Systems, Source Term

1. Introduction

Beginning in 2015, a project was initiated to update and modernize the probabilistic safety assessment (PSA) of the GE-Hitachi Nuclear Energy (GEH) PRISM sodium fast reactor (SFR) [1]. This project is a collaboration between GEH and Argonne National Laboratory (Argonne), and is funded in part by the U.S. Department of Energy. The goal of the project was the development of a next-generation PSA 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. Additionally, this effort was executed in accordance with guidance provided by the recently issued ASME/ANS Non-LWR PRA standard [2], which has been approved for trial use.

This paper provides a summary of the tasks completed by Argonne during the two-year PSA update. An associated paper provides details of the risk insights gained during the project [3]. Section 2 describes the system reliability analysis. This includes both the mechanistic assessment of passive safety system reliability, including the determination of success criteria, and the traditional fault tree assessment of sodium specific components. Section 3 reviews the source term assessment, which encompassed a mechanistic assessment for risk significant event sequences and a simplified analysis for non-risk significant sequences.

2. System Reliability Analysis

This section details the analyses performed to determine system reliability. The first subsection reviews the reliability analysis of passive safety systems, including the RVACS heat removal system and the inherent reactivity feedbacks (IRFs) of the metal fuel core. The second subsection describes the data analysis performed to determine basic failure event probabilities for sodium specific components, such as sodium piping and sodium pumps.

2.1. Passive Systems

The methodology utilized to assess the reliability of the passive safety systems of the PRISM reactor design is detailed elsewhere [4], but an overview of the analysis procedure is shown in Figure 1. As the figure details, the passive system reliability analysis is directly tied to the establishment of success criteria (SC) for the PSA. A summary of the analyses conducted for the RVACS and IRFs is provided here.

2.1.1 RVACS

The RVACS is a safety-grade heat removal system driven by natural circulation that utilizes air from the environment as the ultimate heat sink. The reliability analysis began with the **system identification**. The RVACS includes eight air inlets, which feed into four inlet stacks and then one common inlet plenum. This air then flows down the reactor silo, outside of the collector cylinder. After a 180° turn, the air then rises through a channel between the containment vessel and collector cylinder, before entering a common outlet plenum. From there, the hot air splits into four hot air stacks, and exits the system through four air outlets.

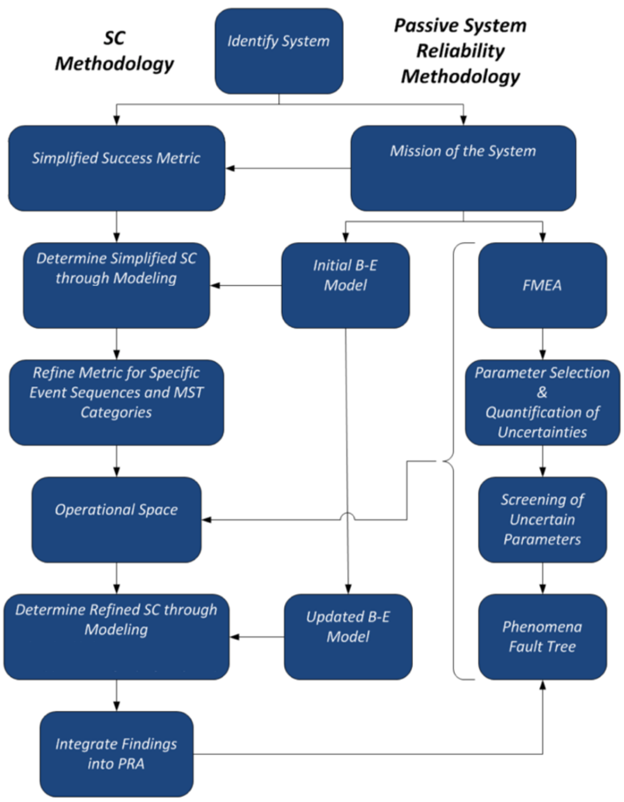

FIG. 1. Passive System Reliability Analysis Procedure

Following system identification, the **system mission** was defined. The central mission of RVACS is to provide removal of all reactor decay and sensible heat following reactor shutdown, when the normal heat rejection paths through the steam generator or through the Auxiliary Cooling System (ACS) are unavailable. The secondary mission is to maintain reactor structural temperatures below safe limits following reactor shutdown in the event that all other heat rejection pathways are unavailable. This mission was utilized to define **simplified success metrics,** which included the prevention of sodium boiling and hot pool temperature thresholds based on cladding failure mechanisms.

Next, a **best-estimate model** was created. This task was performed using the Argonne computer code SAS4A/SASSYS-1 [5]. The SAS4A/SASSYS-1 code has a built-in RVACS model, which has been validated against experiments performed at Argonne [6]. **Simplified SC** were found using the SAS4A/SASSYS-1 model for a variety of protected and unprotected transients.

TABLE I: RVACS – Uncertain Parameters after Screening

Uncertain Parameters
Emissivity
Flow Area
Stack Height
Turbulent Friction Factor
Inlet Pressure Drop Coefficient
Air Temperature

A series of analyses were performed to determine influential factors and associated uncertainties. First, an **FMEA** was performed that established 66 possible failure modes for the RVACS system. The failure modes were grouped into three fundamental categories: flow blockage, flow disruption, and insufficient heat transfer. From there, **parameter selection and uncertainty quantification** was performed for those parameters that were associated with the three fundamental failure modes. Through sensitivity analyses, **screening of the parameters** reduced the parameters to the six shown in Table I. Parameters identified as having negligible effect on the success metrics were precluded from further consideration.

The uncertain parameters were used to determine the **operational space** and were utilized to **refine the SC through additional modeling**. Additional SAS4A/SASSYS-1 simulations were performed for transient scenarios to determine the failure surface (in six-dimensional space, based on the six uncertain parameters). Failure was defined in conjunction with the source term analysis (described in Section 3), which had five fuel damage categories (FDCs) for RVACS transients, shown in Table II. The FDCs establish the extent of fuel damage based on the active fuel batches, the spent fuel stored in the reactor, and vessel integrity. The vessel integrity FDC does not indicate assured vessel failure, but only that additional vessel integrity analysis should be considered.

TABLE II: RVACS - Fuel Damage Categories

FDC
No Damage
Spent Fuel Failure
SF + Batch 4 Fuel Failure
SF + Batch 4,3 Fuel Failure
SF + Batch 4,3,2 Fuel Failure
Vessel Integrity

While the complete results of the RVACS analysis are not provided here due to proprietary restrictions, Figure 2 shows example results for a protected loss-of-flow (PLOF) transient. In total, 729 SAS4A/SASSYS-1 simulations were performed for this transient using a full-factorial sampling method. Each simulation had a different likelihood of occurrence based on the selected values from the six uncertain parameters. Based on the simulations results, a complementary cumulative distribution function (CCDF) could be created, which establishes the probability of violating a FDC threshold for the given accident sequence. For example, the probability of exceeding the "spent fuel damage" FDC threshold temperature for this event sequence was approximately 2E-8. The figure also shows uncertainty bounds placed at ±25K, which were selected based on an preliminary assessment of modeling uncertainties. In general, the inlet pressure drop coefficient was found to be most impactful parameter uncertainty.

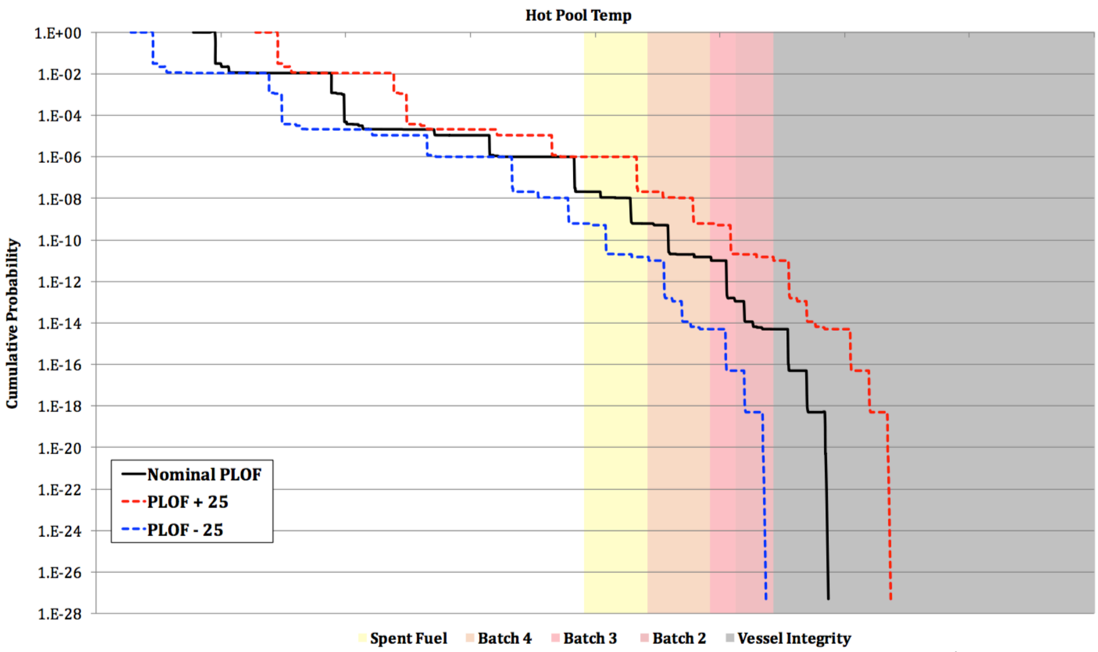

FIG. 2. RVACS - Example CCDF Results for 729 PLOF Simulations[1]

[1] Hot pool temperature values have been removed from the figure due to proprietary restrictions. Coloured sections illustrate FDC thresholds.

2.1.2 Inherent Reactivity Feedbacks (IRFs)

The analysis of the IRFs followed a procedure similar to that of the RVACS analysis, but with several small changes. The IRFs are intrinsic properties of the fuel or core that result in reactivity changes with changes in system temperature. The IRFs are a key feature for metal fuel, pool-type SFR designs, such as PRISM.

The **system identification** of the IRFs included not only the intrinsic reactivity properties of the reactor system, but also the Gas Expansion Modules (GEMs), which are an engineered system developed to lower reactivity in loss-of-primary-flow transients. The **mission of the system** is to bring the reactor to a new steady-state condition while still critical, but at a lower power and higher temperature during unprotected transients. This power level should be safely sustainable until ultimate shutdown can be achieved. The mission was utilized to define **simplified success metrics,** which included cladding failure thresholds and fuel melting.

As with the RVAC analysis, a **best-estimate model** was developed using SAS4A/SASSYS-1, which includes internal models for the IRFs under consideration. The values for the IRFs were calculated separately through detailed neutronic analyses. **Simplified SC** were found using the SAS4A/SASSYS-1 model for several unprotected transients.

Unlike the RVACS analysis, an FMEA was not performed, as separate, detailed analyses would be necessary for each IRF, which was outside the scope of the current project. However, a previous analysis was used as the basis to establish the **parameter selection and quantification of uncertainties**. The IRF parameters and their associated uncertainties, can be found in Table III. There was no need to **screen uncertain parameters**, as there were only seven IRFs under consideration.

TABLE III: IRFs – Uncertain Parameters

Reactivity Feedback Mechanisms	1σ Uncertainty
Doppler	20%
Sodium Density	20%
Fuel Axial Expansion/Contraction	30%
Net Radial Expansion	50%
Control Rod Expansion (Neutronic)	10%
Control Rod Expansion (Thermal Hydraulic)	20%
GEMS (for loss-of-primary-flow transients)	~20% (one-sided normal distribution)

TABLE IV: IRFs - Fuel Damage Categories

Fuel Damage Category
No Damage
Batch 4 Fuel Failure
Batch 4,3 Fuel Failure
Batch 4,3,2 Fuel Failure
Batch 4,3,2,1 Fuel Failure
Sodium Boiling/Fuel Melting

The uncertain parameters were used to define the **operational space**, which was used to **refine the SC through additional modeling**. Additional SAS4A/SASSYS-1 simulations were performed for unprotected transient scenarios to determine the extent of core damage. Core damage thresholds were established in conjunction with the IRF FDCs of the source term assessment. Six FDCs were utilized, shown in Table IV, which assess the number of fuel pin failures (by active core fuel batch) and the onset of sodium boiling or fuel melting. Since the SAS4A/SASSYS-1 simulations for the IRFs were fast-running when compared to the RVACS transients (due to a much shorter mission time), Monte Carlo sampling was utilized in place of a full-factorial experiment design.

While the complete results of the IRF analysis are not provided here due to proprietary restrictions, Figure 3 shows example results for an unprotected loss-of-flow (ULOF) transient. In total, 5000 SAS4A/SASSYS-1 simulations were performed for the ULOF transient. The figure shows the distribution of the 5000 peak fuel temperatures based on the normalized values of the six uncertain parameters. As to be expected, for the ULOF transient, the response of the

GEMs is particularly influential. Unlike the RVACS analysis, which utilized a peak coolant temperature to determine the FDC, the IRF analysis used the internal SAS4A/SASSYS-1 fuel damage models to assess the extent of fuel damage for each accident scenario. Since Monte Carlo sampling was utilized, each of the 5000 simulations were of equal probability. Therefore, FDC probabilities could be established by taking the total number of simulations that entered a FDC divided by the total number of simulations (additional statistical confidence intervals were also utilized).

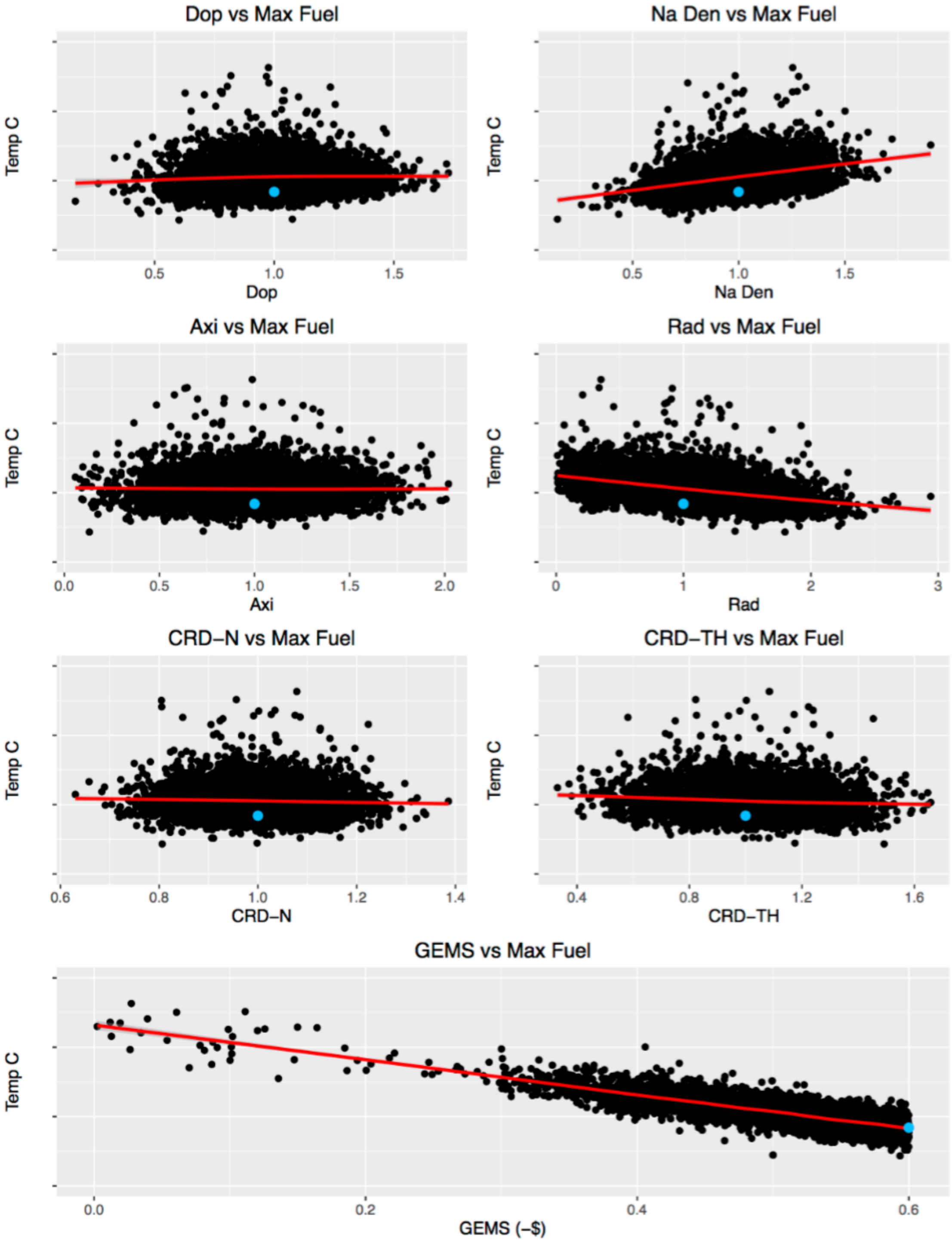

FIG. 3. IRFs - Example Results for 5000 ULOF Simulations[2]

[2] Temperature values have been removed from the figure due to proprietary restrictions. Blue dots indicate simulation result utilizing nominal values.

2.2.Sodium Specific Components

For components not associated with passive safety systems, traditional fault tree analyses were performed. To aid in the fault tree assessments, conducted by GEH, Argonne established basic failure event probabilities for sodium specific components. This included sodium pumps, sodium piping, sodium heat exchangers (intermediate heat exchanger and steam generator), and sodium valves. The methodology utilized to determine the basic failure event probabilities is detailed elsewhere [7], but an overview is shown in Figure 4. This section provides a brief review of the process.

The first step of the process was to **identify the system and failure events under consideration**. This was conducted by GEH as part of their initiating event and system analysis. By far, the largest part of the effort was the **review of available data sources**. This included U.S. SFR component databases (CREDO), raw data from SFR plant logs, non-SFR nuclear databases, and non-nuclear data. This data was compared through a relevancy test.

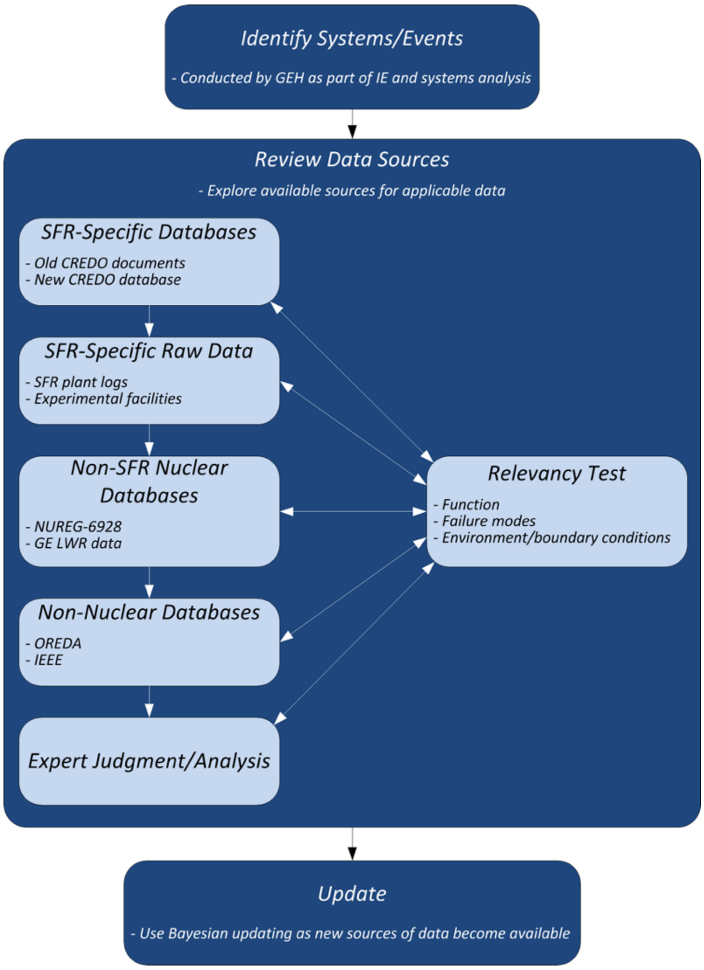

FIG. 4. Basic Failure Event Probability Analysis Procedure

In general, the U.S. SFR component database and SFR raw data were the most valuable sources for determining basic failure event probabilities. However, since much of the SFR data was older than 20+ years, the LWR data provided an important point of comparison for the SFR findings. For example, by comparing the component conditions, such as temperature, pressure, etc., a general judgement could be made regarding whether SFR components would likely be more or less reliable than similar LWR components. If an initial reliability analysis finding did not match these expectations, more research was performed to investigate possible causes.

The **expert judgment/analysis** step of the data review was necessary for each component examined, as data was consolidated from multiple sources. The expert analysis was necessary to weight the applicability of the data sources (using the relevancy test). In some cases, such as with the sodium piping and intermediate heat exchanger, structural analyses were combined with past data to provide updated basic failure event probabilities. Throughout the process, it was important to properly document the assumptions and reasoning associated with the techniques to combine data from multiple sources.

Due to proprietary and data access limitations, detailed findings from the basic failure event analysis are not provided here. However, in general, the greatest difficulties encountered during the analyses related to the proper use of past reliability data for components that were smaller in size or capability than those to be used in PRISM (as the U.S. has not previously operated an SFR of size similar to PRISM). Therefore, testing data on prototypic PRISM components was important in updating the reliability data from smaller components. Additionally, there were challenges regarding how to properly credit improvements in materials and component design since the construction of past U.S. SFRs, such as EBR-II and FFTF. For example, several past component failures were considered no longer applicable (as determined through the relevancy test), as the component design flaw that caused the failure has since been rectified.

3. Source Term Analysis

The evaluation of the source term was a key segment of the PRISM PSA update/modernization. The methodology utilized for the source term assessment is detailed elsewhere [8], but follows the procedure outlined in Figure 5. This section provides an overview of the analysis and results. The first subsection reviews the mechanistic source term assessment for risk-significant sequences (RSSs), while the second subsection reviews the source term assessment for non-RSSs with the possibility of radionuclide release.

3.1. Risk Significant Sequences

Following an initial event sequence (or FDC) assessment and categorization, a mechanistic source term (MST) assessment was performed for those event sequences that were determined to be risk significant (or which may be risk significant). Due to the constraints of the project, the mechanistic source term assessment was a simplified version of the analysis detailed in ref [9] and ref [10] .

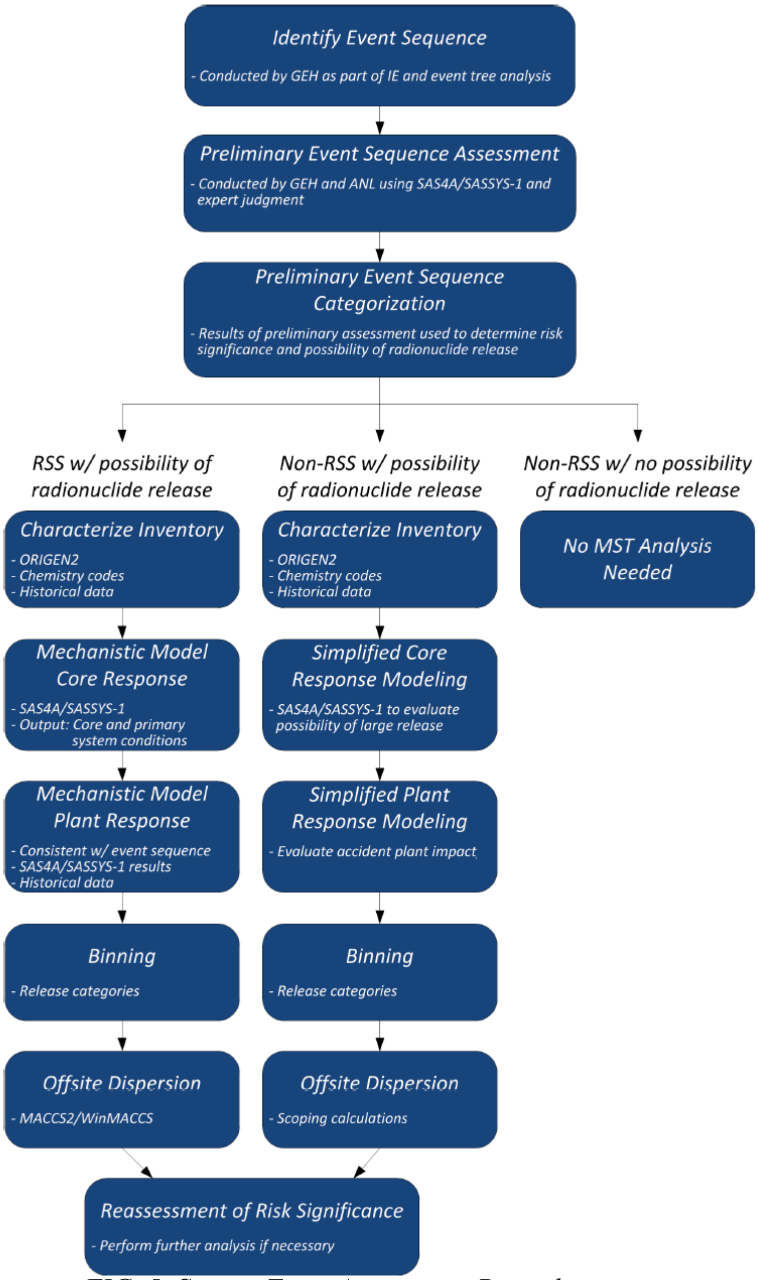

FIG. 5. Source Term Assessment Procedure

The MST assessment began with a **characterization of the core inventory**. As all transient scenarios were assumed to occur at the same time-in-cycle, this step was only performed once. It included ORIGEN [11] calculations to determine radionuclide inventory and a review of historic data [12] and the use of chemistry codes [13] to assess the migration of radionuclides within the fuel pin to the fission gas plenum or bond sodium. The four fuel batches were assumed to be at different burnup levels, which changes the radionuclide migration fractions.

Next, the output of the SAS4A/SASSYS-1 simulations (**the mechanistic core response**), which were conducted as part of the SC assessment, were analyzed to determine the extent of fuel damage. This step coincides with the establishment of FDCs, discussed in Section 2. The

SAS4A/SASSYS-1 results not only provided the extent of core damage, but also the conditions of the primary system (temperature/pressure).

The **mechanistic model of the plant response** was performed next, and included the five phases shown in Figure 6. For the current project, a simplified radionuclide transport code was developed to simulation radionuclide movement through the five phases. First, radionuclide release from the failed fuel pins to the primary sodium was determined based on the conditions of the fuel pin at the time of failure and utilizing past experimentation and data [12]. Next, radionuclide release from the sodium pool to the cover gas region was assessed. This phase included both radionuclide transport within fission gas bubbles and radionuclide vaporization from the sodium pool. The release of radionuclides from the cover gas region to the containment was simulated next, using a preassigned reactor head leakage rate. A similar process was used for radionuclide release from the containment to environment. In both the cover gas region and containment, the deposition of radionuclide aerosols was considered as a potential removal pathway. Each RSS event sequence analysis also explored the possibility of radionuclide barrier (reactor head, containment, etc.) bypass routes. The probability of barrier bypass was dependent on the event sequence under consideration. Lastly, the **offsite dose calculations** (which were conducted within this step of the analysis, as **binning** was not necessary) were performed using WinMACCS [14].

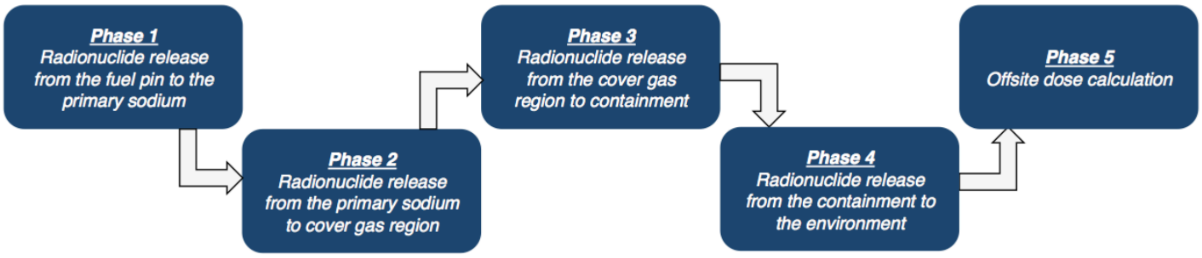

FIG. 6. Five Phases of Mechanistic Source Term Assessment

As with the results of the reliability assessment presented in Section 2, complete source term results are not included here due to proprietary restrictions. However, an example result is shown in Figure 7 for a long-term degraded heat removal transient, which results in damage to a large fraction of the core inventory of fuel pins. As the figure shows, the releases are separated by analysis phase to aid in the determination of important transport and retention mechanisms. In this event sequence, noble gases dominated the released activity, as other radionuclide elements were largely retained within the fuel or primary sodium. This was true for the majority of RSSs.

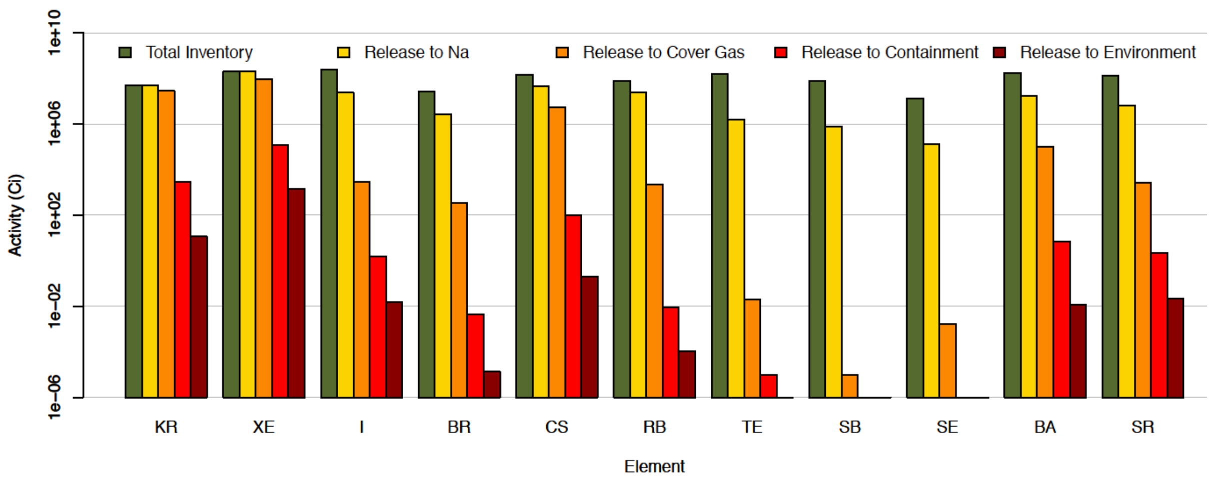

FIG. 7. Example MST Analysis Results

3.2. Non-Risk Significant Sequences

For those event sequences which were initially categorized as non-risk significant (non-RSSs), but with a possibility of radionuclide release, a simplified source term analysis was performed to evaluate the potential for large radionuclide releases. In general, the non-RSSs were very low frequency sequences, with multiple failures impacting multiple radionuclide barriers.

The non-RSS assessment procedure also starts with the **characterization of the core inventory**. Again, the same core inventory was used for all event sequences, since they were assumed to occur at the same time-in-cycle. Next, the **core response modeling** was completed using SAS4A/SASSYS-1. For the non-RSSs, the focus of the SAS4A/SASSYS-1 simulations was the possibility of large radionuclide releases. For example, attention was paid to the integrity of the primary and guard vessel, sodium boiling, and bulk fuel melting.

The **simplified plant response modeling** focused on the impact of the radionuclide barriers. For example, if vessel failure was a possibility for the event sequence, the source term analysis was modified to include radionuclide release directly to the spacing between the primary vessel and guard vessel (or directly to the environment through RVACS in the unlikely scenario where failure of both vessels occurred).

Scoping calculations were performed for the **offsite dispersion** analysis. The main goal of the offsite dispersion analysis was to determine the approximate order of magnitude of the offsite consequence. These results were used to **reassess the risk significance** of the event sequence. For example, even though the event sequences originally considered non-RSS were of very low frequency, a very large offsite consequence could result in the event sequence actually being risk significant. If this was the case, the event sequence was then recategorized and the mechanistic source term assessment was then performed.

For the analysis described here, the assessment of non-RSSs (which included event sequences such as double vessel – primary and guard – failure) involved many sensitivity analyses, as there were large uncertainties associated with such low frequency sequences. Sensitivity analyses examined the impact of the assumptions associated with phenomena such as sodium boiling (radionuclide vaporization from boiling sodium), radionuclide release fractions from high temperature molten fuel, and radionuclide release from burning sodium. None of the non-RSSs were found to be risk significant after the simplified source term assessment was completed, as the offsite consequences were not large enough to outweigh the incredibly low frequency of occurrence.

4. Conclusions

As part of a project to update and modernize the PRISM SFR PSA, Argonne conducted system reliability analyses and a source term assessment. The system reliability analysis consisted of both mechanistic analyses of passive safety system performance and the development of basic failure event probabilities for sodium specific components that would be assessed using traditional fault tree analysis. The passive system assessment utilized mechanistic modeling to develop success criteria that were consistent with the fuel damage categories of the source term analysis.

The source term analysis was divided into mechanistic and simplified analyses, depending on the risk significance of the event sequence (or fuel damage category). A computer code was developed for the mechanistic assessment, which determined the transport and retention of radionuclides through each of the five stages of release. The simplified analysis focused on the possibility of large radionuclide releases for extremely rare event sequences, and relied heavily on sensitivity analyses to explore the bounds of radionuclide release due to large uncertainties.

Additional information regarding the PRISM PSA update/modernization, and risk insights from the completed PSA results, can be found in ref [3].

5. Acknowledgements

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