ANL-ART-41

Regulatory Technology Development Plan Sodium Fast Reactor

Mechanistic Source Term – Trial Calculation: Work Plan

Nuclear Engineering Division

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1 Introduction

The development of a mechanistic source term (MST) will be a vital element of the licensing process for U.S. advanced reactors. In FY13, an investigation by Argonne National Laboratory (Argonne) regarding sodium fast reactor (SFR) licensing gaps identified the MST as a possible impediment to future licensing [1]. Following this work, a project was initiated to characterize the current state-of-knowledge regarding SFR MSTs [2]. The findings of this study indicated that a deficiency may exist regarding the ability of current computational tools to properly assess the release and transport of radionuclides during an SFR core damage accident. It was suggested that a trial MST calculation be performed to evaluate the capabilities of the current suite of tools to inform future research and development efforts. The project outlined here seeks to perform this task.

1.1 Objective

The overall objective of the SFR Regulatory Technology Development Plan (RTDP) effort is to identify and address potential impediments to the SFR regulatory licensing process. In FY14, an analysis by Argonne [1] identified the development of an SFR-specific MST methodology as an existing licensing gap with high regulatory importance and a potentially long lead-time to closure. This work was followed by an initial examination of the current state-of-knowledge regarding SFR source term development (ANL-ART-3 [2]), which reported several potential gaps. Among these were the potential inadequacies of current computational tools to properly model and assess the transport and retention of radionuclides during a metal fuel pool-type SFR core damage incident[1].

The objective of the current work is to determine the adequacy of existing computational tools, and the associated knowledge database, for the calculation of an SFR MST. To accomplish this task, a trial MST calculation will be performed using available computational tools to establish their limitations with regard to relevant radionuclide release/retention/transport phenomena. The application of existing modeling tools will provide a definitive test to assess their suitability for an SFR MST calculation, while also identifying potential gaps in the current knowledgebase and providing insight into open issues regarding regulatory criteria/requirements. The findings of this analysis will assist in determining future research and development needs.

In addition to the examination of existing computational tools, the trial MST calculation will also determine the transport/retention phenomena that are most impactful on the resulting offsite consequence. It is vital that adequate computational tools exist for the assessment of the important phenomena identified through this process. The information resulting from this activity will assist in the prioritization of future research and development efforts, if gaps are identified. Additionally, the results of the phenomena assessment may demonstrate that certain transport/retention mechanisms can be analyzed utilizing less sophisticated modeling approaches, such as the use of correlations developed from existing empirical data, if it can be shown that the impact of such an approach on the resulting offsite consequence is minor.

The trial MST analysis approach will be developed with input from industry SFR developers, to ensure consistency with proposed industry SFR designs and source term analysis methodologies. The analysis should not only address the adequacy of existing tools for the determination of offsite dose during accident sequences, but also the magnitude of land contamination, as the latter metric has received increased attention following the events at the Fukushima Daiichi nuclear power plant.

[1] Another gap noted in ANL-ART-3, regarding the magnitude of radionuclide release following the failure of metal fuel in liquid sodium, was recently investigated in ANL-ART-38 [7].

1.2 Scope

A trial MST calculation will be performed for a metal-fuel, pool-type SFR core damage accident. This task involves assessing the consequences of a reactor transient in terms of the magnitude and characteristics of fuel failure and the resulting release and transport of radionuclides. There exists a wide variety of phenomena that affect the transport and retention of radionuclides released from metal fuel in a pool-type SFR. The trial MST calculation will attempt to adequately model and assess these phenomena using a variety of available computational tools and past experimental/accident findings.

Despite several past statements encouraging the use of an MST analysis for advanced reactor licensing [3,4], the NRC currently has no formal MST requirements. Therefore, past NRC staff documents and recent NRC interactions with the NGNP Project [5] will provide the basis for determining the adequacy of the MST analysis. Additional insight will be provided by past and current NRC staff, when possible.

The trial MST analysis will focus on the consequences of a bounding beyond design-basis core damage accident for a metal fuel, pool-type SFR[2]. Accidents within this category may result in fuel pin failures or limited fuel melting, but a coolable core geometry, vessel integrity, and core submersion in sodium are maintained. Core and plant conditions which are typical of accidents in the residual risk category or beyond[3], such as core energetics or primary vessel failure, will not be examined. The reactor design under investigation will be a pool-type SFR, utilizing a ternary U-Pu-Zr sodium-bonded metal fuel. The plant design and characteristics will be chosen to be representative of current industry designs, although no specific features of proprietary designs will be utilized.

[2] Historically, a detailed source term analysis has not been conducted for beyond design-basis accidents during the licensing process. However, for metal fuel, pool-type SFRs, core damage accidents typically do not occur within the design-basis accident category, therefore the focus of the trial MST calculation will be on a beyond design basis accident scenario.

[3] Typically, accidents within this range are practically eliminated and detailed analysis is not necessary.

2 Proposed Methodology

The trial MST calculation will be performed using parallel methodologies, as shown in Figure 2–1. First, a mechanistic calculation, using available best-estimate tools and models, will attempt to realistically characterize the radionuclide release and offsite consequences of an SFR core damage accident. This calculation includes the analysis of uncertainties and their impact on the resulting metrics. In contrast to the mechanistic analysis, an independent simplified sensitivity calculation will be performed to determine the impact of various source term analysis factors on offsite consequences. The results of the sensitivity calculation will assist in prioritizing gaps found during the mechanistic analysis, and also provide bounding information on the offsite consequences.

Mechanistic Calculation

- *Attempt to realistically model radionuclide release & transport*
- *Use available best-estimate codes and models*
- *Identify gaps in current knowledgebase or computational tool capability*

Sensitivity Calculation

- *Use simplified calculations to conduct parametric evaluation*
- *Determine the importance of radionuclides and retention mechanisms*
- *Assist in the prioritization of gaps identified in mechanistic calculation*

Figure 2–1: Trial Source Term Analysis Methodologies

The transport of radionuclides during a beyond design-basis SFR core damage accident can be divided into several primary phases, as shown in Figure 2–2. The accident begins with the reactor transient and subsequent damage to fuel within the core. This is followed by the release of radionuclides from the damaged fuel, and their transport/retention in the primary sodium pool. From there, some radionuclides may migrate to the cover gas region, and subsequently to containment (assuming containment bypass pathways are not available), and can be released to the environment. The final phase is the dispersion of radionuclides from the reactor site.

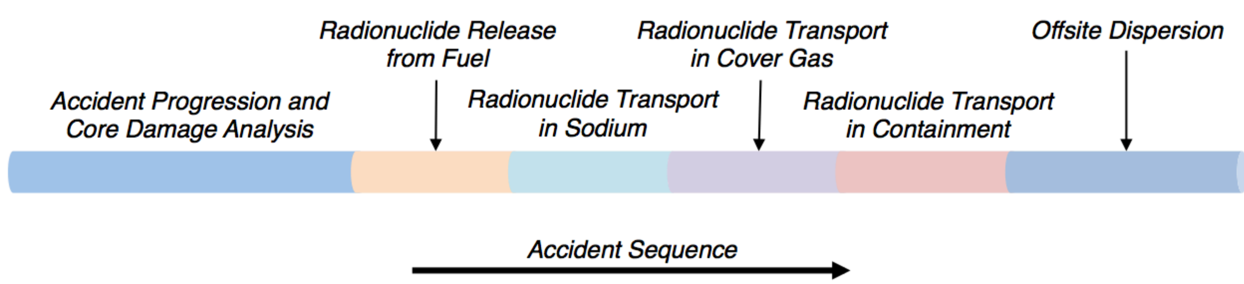
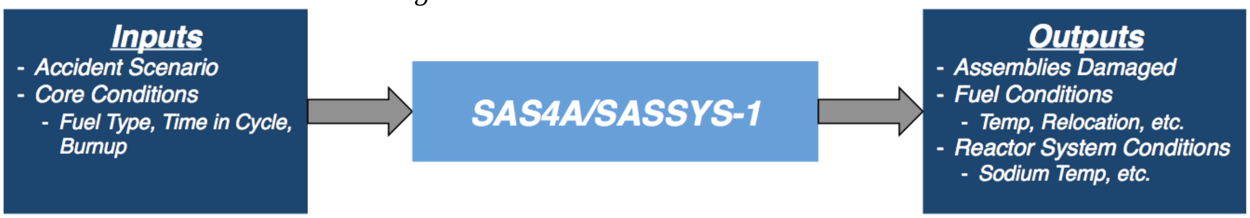


Figure 2–2: Source Term Analysis Sequence

2.1 Mechanistic Calculation

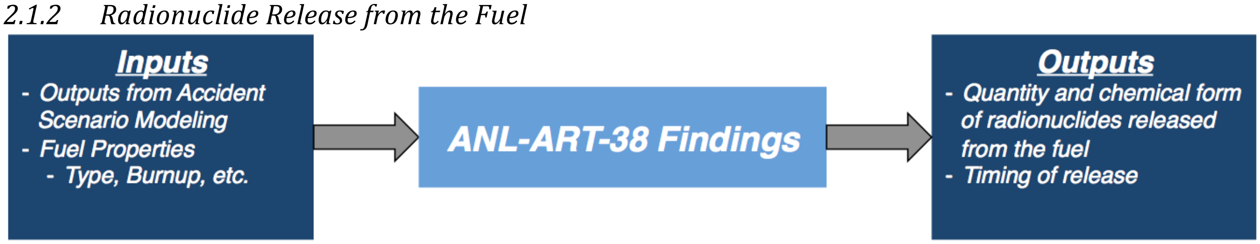
This section outlines the proposed methodology for the mechanistic calculation. Each subsection reviews a single step of the source term analysis sequence shown in Figure 2–2, including the inputs and outputs of each phase.

2.1.1 Accident Scenario Modeling



The MST analysis begins with the modeling of the accident sequence using SAS4A/SASSYS-1[4,5] [6]. The inputs to the calculation are the accident scenario to be modeled and the core conditions, including the fuel type under consideration, the time in cycle of the accident, and the burnup level of the fuel. SAS4A/SASSYS-1 is capable of modeling the progression of the accident sequence, including the impact on fuel integrity. The outputs of the analysis include the number of assemblies (or fuel pins) damaged, including the extent of fuel damage (cladding failure, fuel melting, fuel relocation, etc.), and timing of the damage. SAS4A/SASSYS-1 also provides information regarding the state of the reactor system, such as the temperature and flow rate of the primary sodium, and the temperature of the reactor vessel and cover gas region.

2.1.2 Radionuclide Release from the Fuel



The next step of the MST analysis investigates the release of radionuclides from the damaged fuel pins into the primary sodium. For this analysis, the outputs of the previous steps are utilized, including the timing and extent of fuel damage. Additional information regarding the properties of the fuel, including the specific fuel type, burnup level, and radionuclide inventory, is also needed. As noted in Section 1.1, a recent project conducted by Argonne examined radionuclide partitioning within the fuel pin and the release of radionuclides from damaged metal fuel pins at various temperature conditions (ANL-ART-38 [7]). The findings of this project will serve as the basis of the radionuclide release estimates, as there are currently no computational tools capable of determining radionuclide release from failed sodium-bonded metal fuel pins. The outputs of this MST analysis step include the quantity and chemical form of the radionuclides released from the fuel. Additional information regarding the timing of the radionuclide released may also be provided, as some radionuclides may not be released at the moment of fuel pin failure.

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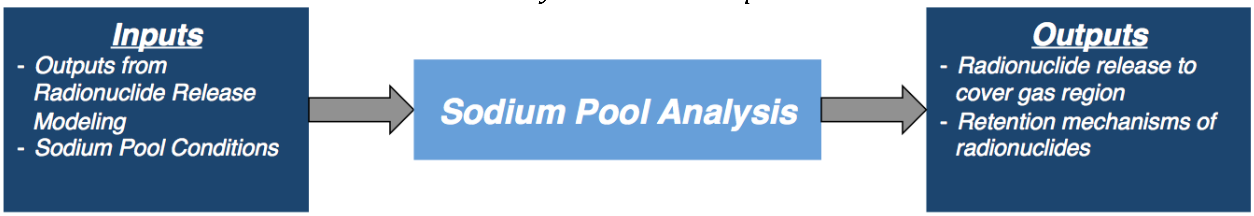
4 Currently, efforts at Argonne are exploring the use of SAS4A/SASSYS-1 as part of SFR regulatory licensing applications.

5 Other reactor transient analysis codes will also be assessed to determine their capabilities with regard to modeling the failure of metal fuel in an SFR.

2.1.3 Radionuclide Behavior in the Primary Sodium and Vaporization

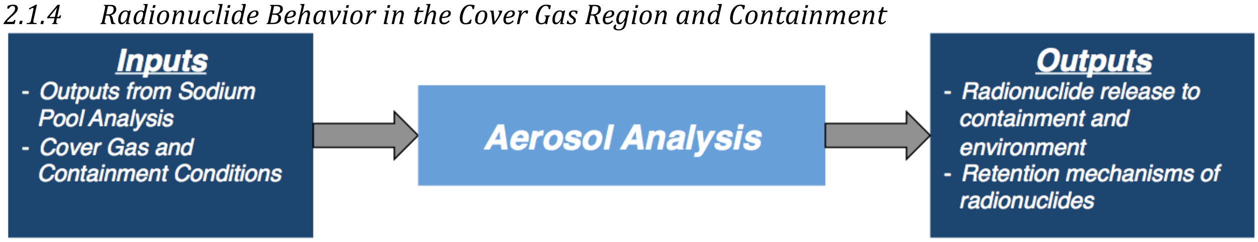


The third MST analysis step examines the behavior of radionuclides in the primary sodium pool. This analysis requires the outputs of the previous MST analysis step (the quantity and chemical form of radionuclides released from the fuel) and the conditions of the sodium pool (determined by SAS4A/SASSYS-1 in Section 2.1.1). The sodium pool analysis is an agglomeration of several modeling approaches, as no single sodium pool analysis tool currently exists. First, chemical thermodynamic equilibrium codes, such as HSC Chemistry [8] or Thermo-Calc [9], will be used to determine the behavior of the radionuclides in liquid sodium. These tools are useful for predicting the compounds that will develop within the sodium, which is important in determining the likelihood of vaporization to the cover gas region.

There are other factors that must also be considered as part of the sodium pool analysis. First, the release of noble gases from the fuel pin will result in the creation of bubbles in the primary sodium. It is possible that other radionuclides can be transported within these bubbles to the sodium-cover gas interface. Efforts to predict and model the bubble transport mechanism developed during the Integral Fast Reactor project will be leveraged in an attempt to account for this phenomenon. Second, many radionuclides may deposit within the primary sodium system, whether due to gravitational settling of radionuclide bearing particulates or adsorption onto metal surfaces. This retention mechanism can limit the quantity of some radionuclides available for vaporization. However, past experimental evidence is the primary source of data, as computational tools to predict this behavior do not currently exist.

Radionuclide transport from the sodium pool to the cover gas region is primarily accomplished through vaporization. Predictive modeling of the evaporative releases of radionuclides requires quantification of: (1) radionuclide speciation (i.e. the distribution of an element among its possible molecular forms), (2) the vapor pressures of the dominant radionuclide species, and (3) mass transport (e.g. by convection) in the cover gas that moves vaporized radionuclides away from the sodium pool surface. The thermodynamic codes HSC Chemistry and Thermo-Calc will be used to quantify the radionuclide speciation based on Gibbs free energy minimization techniques and will be used to calculate the species vapor pressures based on activity coefficient models that will be developed and implemented for this study.

2.1.4 Radionuclide Behavior in the Cover Gas Region and Containment



The next step of the MST analysis[6] investigates the transport and retention of radionuclides in the cover gas region and containment. This analysis will utilize outputs from the previous MST analysis step, which includes the quantities and chemical forms of the radionuclides from the sodium pool analysis. The conditions of the cover gas region (temperature and pressure) will be derived from the SAS4A/SASSYS-1

[6] Performed in conjunction with Sandia National Laboratories.

simulation. The analysis will be preformed utilizing CONTAIN2-LMR to track the transport of radionuclide aerosols from the sodium pool surface, through the cover gas region into containment, and finally to the environment. Sensitivity studies will be performed to determine the effects of reactor head and containment leak rates as well as the impact of the size of the containment structure on retention and transport. The outputs from this step of the analysis will be the quantities and chemical forms of radionuclides released from the cover gas region into containment and from the containment to the environment.

2.1.5 Offsite Radionuclide Dispersion

In the final step of the MST analysis[6], the results of the aerosol analysis, along with siting data, will be utilized to quantify offsite dose and land contamination. These outputs will be determined by performing atmospheric transport analysis using the WinMACCS code [10]. Sensitivity analyses will be performed to investigate the effects of meteorology and containment leak rate on the output variables (early/latent health effects and land contamination/economic impacts).

2.2 Independent Sensitivity Analysis

In parallel to the mechanistic analysis, an independent sensitivity calculation will be performed to identify the radionuclides and transport/retention mechanisms of highest importance to offsite dose and possible land contamination. This calculation does not involve mechanistic radionuclide release and transport computer codes, but uses simplified assumptions to explore the possible range of radionuclide release. The sensitivity calculation will be performed by an independent contractor[7], as a mechanism to ensure that any potential assumptions or biases of the staff performing the mechanistic calculation are not also present in the sensitivity analysis. To assist in the examination of multiple sensitivity analyses, the RASCAL radionuclide offsite dispersion code, which was developed by the NRC for rapid assessments [11], will be utilized.

The sensitivity calculation serves several functions. First, the calculations provide insight into which radionuclides are important for offsite dose and possible land contamination. While the radionuclides of importance are well known for light water reactors, it is less certain which elements or isotopes are will be dominant for realistic metal fuel SFR releases (especially in terms of possible land contamination, which has historically not been a major focus of the licensing process). This information will assist in determining which transport and retention mechanisms are of the greatest significance in the SFR MST calculation. If gaps in the knowledgebase or computational capabilities are discovered during the mechanistic calculation, the findings of the sensitivity analysis will assist in prioritizing future work.

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[7] Dr. Richard Denning (ret. Ohio State University, Battelle, U.S. Nuclear Regulatory Commission) will conduct the independent sensitivity calculation. Dr. Denning is a recognized expert in the fields of nuclear safety and severe accident behavior. He served as a technical expert in the area of fission product behavior for the NUREG-1150 light water reactor analysis, and developed methods for the analysis of metal fuel SFR severe accident behavior as part of a NERI project with the Massachusetts Institute of Technology.

3 Timeline

The project timeline can be seen in Figure 3–1. The project work plan was released for review and comment in mid-January, with a final version published in February. A mid-project status report will be released in June, with a draft of the final report released for review in September. The final project report will be published (open access) in mid-October.

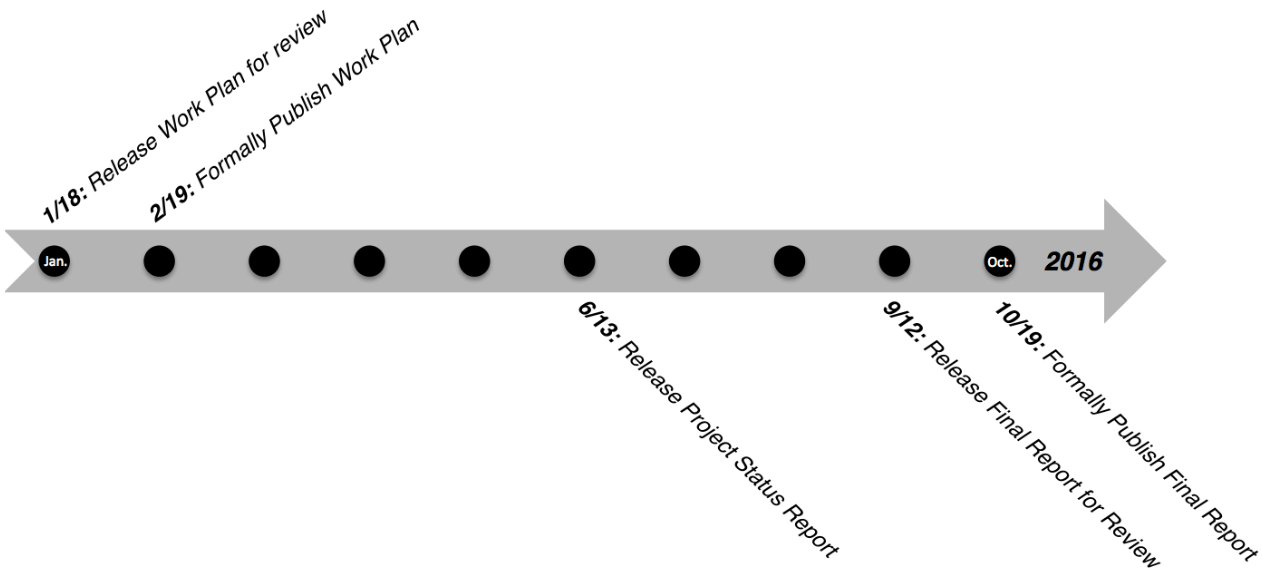

Figure 3–1: Project Timeline

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