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prepared by
David Grabaskas, Matthew Bucknor, James Jerden, Richard Denning (Consultant)

Nuclear Engineering Division
Argonne National Laboratory
9700 South Cass Avenue, Bldg. 208
Argonne, IL 60439-4854

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A Mechanistic Source Term Calculation for a Metal Fuel Sodium Fast Reactor

D. Grabaskas[1], M. Bucknor[1], J. Jerden[1], R. Denning[2]

[1]Argonne National Laboratory, Argonne, Illinois, USA
[2]Consultant, Columbus, Ohio, USA

E-mail contact of main author: dgrabaskas@anl.gov

Abstract. A mechanistic source term (MST) calculation attempts to realistically assess the transport and release of radionuclides from a reactor system to the environment during a specific accident sequence. The U.S. Nuclear Regulatory Commission (NRC) has repeatedly stated its expectation that advanced reactor vendors will utilize an MST during the U.S. reactor licensing process. As part of a project to examine possible impediments to sodium fast reactor (SFR) licensing in the U.S., an analysis was conducted regarding the current capabilities to perform an MST for a metal fuel SFR. The purpose of the project was to identify and prioritize any gaps in current computational tools, and the associated database, for the accurate assessment of an MST. The results of the study demonstrate that an SFR MST is possible with current tools and data, but several gaps exist that may lead to possibly unacceptable levels of uncertainty, depending on the goals of the MST analysis.

Key Words: Sodium Fast Reactor, Source Term, Metal Fuel, Safety Analysis

1. Introduction

The U.S. Nuclear Regulatory Commission (U.S. NRC) has repeatedly indicated an expectation that future advanced reactor license applications should include a mechanistic assessment of potential radionuclide release [1-3]. An Argonne National Laboratory (Argonne) study found that the development of a mechanistic source term (MST) for metal fuelled sodium fast reactors (SFRs) was a potential impediment to licensing efforts in the U.S. To address this subject, Argonne initiated a project to develop a metal fuelled, pool-type SFR MST. This paper provides an overview of one portion of this project, a trial MST calculation. Additional detail regarding the motivation for the project, the data and models utilized, and the results and conclusions can be found in ref [4].

2. Background

Figure 1 shows a high-level MST development pathway. The first two steps of the SFR MST development were completed previously, and are summarized in Argonne report ANL-ART-3 [5]. ANL-ART-3 identified and characterized radionuclide sources within the plant and potential transport and release phenomena. ANL-ART-3 also identified two main gap areas:

1) ***Metal Fuel Radionuclide Release Fractions*** – It was unclear whether sufficient data and experience existed to develop radionuclide release fractions from failed metal fuel pins to the primary sodium.
2) ***Trial MST Calculation*** – No single, comprehensive computer code existed for an SFR MST. Instead, there were several computational tools that could analyse separate parts of the calculation. It was recommended to conduct a trial MST to determine the sufficiency of this approach and the adequacy of the related computational tools.

Both gaps relate to the third step of the MST development pathway, which is the modelling of radionuclide transport phenomena. The first gap was addressed in ANL-ART-38 [6], which attempted to estimate realistic radionuclide release fractions from failed metal fuel pins based on past data. The second gap, regarding the sufficiency of current modelling tools and data, is addressed in the current work.

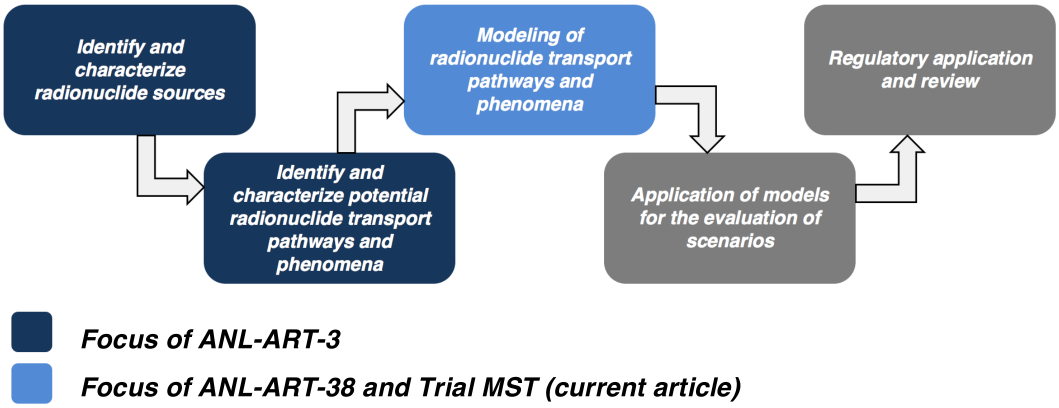

FIG. 1. MST Development Pathway and Project Reports

3. Methodology

The goal of the trial MST calculation is to gauge the adequacy of current computational tools and the associated knowledge base for the assessment of an SFR MST. To accomplish this goal, a parallel-path analysis methodology was developed, as shown in Figure 2. The parallel paths consist of both an MST calculation and a simplified source term sensitivity analysis. The MST calculation uses best-estimate codes and models in an attempt to realistically assess the transport and release of radionuclides. The goal of this analysis pathway is to identify any gaps in current computational capabilities or the associated knowledge base. The sensitivity calculation uses simplified radionuclide transport and retention models to conduct a parametric evaluation. The purpose of the sensitivity calculation is to rank transport and retention phenomena in terms of their importance on offsite consequences (offsite dose and land contamination). Taking the results of both analyses together, a research prioritization plan can be developed.

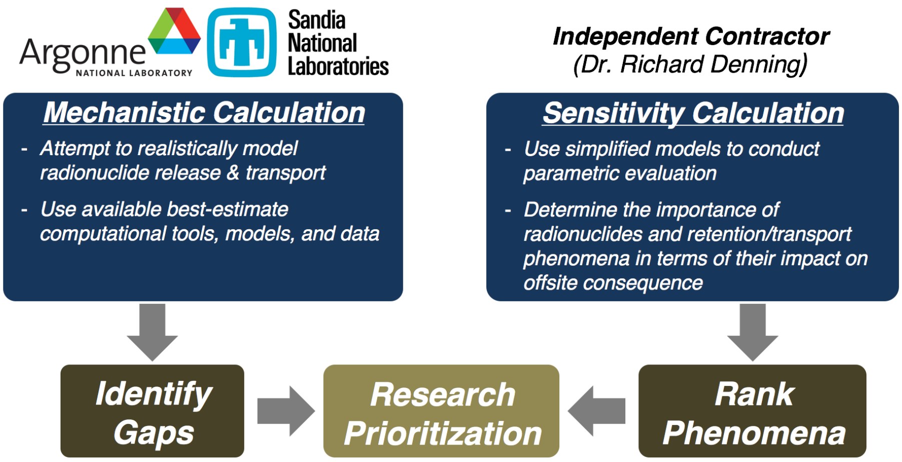

FIG. 2. Project Methodology

Section 4 of this report provides an overview of the trial MST calculation, while Section 5 provides a summary of the sensitivity calculation. Section 6 then details the overall project findings and research prioritization.

4. Mechanistic Source Term Calculation

The trial MST calculation followed the procedure outlined in Figure 3. Each of the analysis steps is briefly reviewed in the followed subsections, along with a description of the reference reactor design and transients assessed.

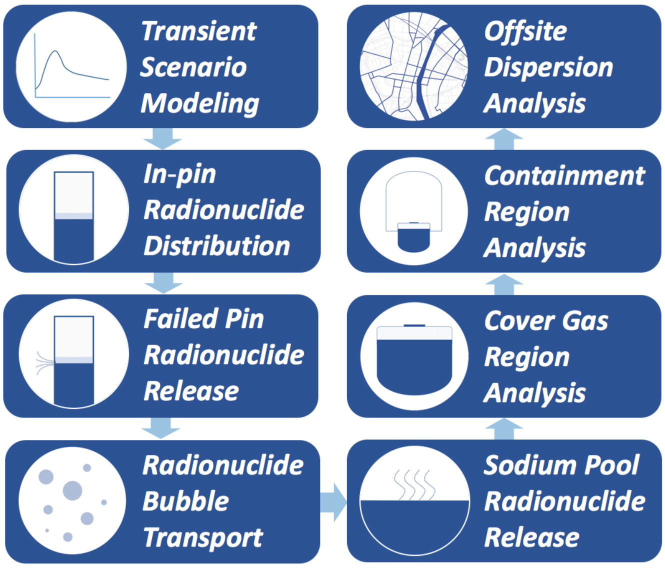

FIG. 3. Trial MST Calculation Procedure

4.1. Reference Reactor

To complete the trial MST calculation, a reference SFR design was created that is similar to the designs currently proposed by U.S. industry. Table I shows the reference reactor is a 1000 MWth, metal fueled, pool-type SFR. It was assumed that the core contained three main fuel batches at different burnup levels, 2at.%, 5at.%, and 10at.%.

TABLE I: Reference Reactor Design

Parameter	Description
Reactor Power	1000 MWth/380 MWe
Type	Pool-type
Purpose	Actinide Burner
Core Fuel	U-Pu-Zr
Cladding Material	HT-9
Core Fuel Batches	3
Primary Coolant Temp – Inlet/Outlet	350°/500°C
Number of Fuel Assemblies	180
Fuel Pins per Assembly	271

4.2. Transient Scenarios and Modeling

As the goal of the trial MST was to gauge the adequacy of current modeling tools and data, transient scenarios were assessed to identify those that would incorporate all pertinent phenomena regarding radionuclide transport and retention. Two transient scenarios were selected with differing transient conditions:

1) ***PLOF+*[1]*:*** A protected loss-of-flow and loss-of-heat-sink coupled with severely degraded decay heat removal (DRACS) capabilities. This transient results in a long, slow heat-up of the core and primary system, with fuel pin failures due to high hoop stress caused by eutectic penetration of the cladding (with no fuel melting) and very high primary sodium temperatures.

2) ***UTOP+*[2]*:*** An large unprotected transient overpower coupled with degraded radial negative reactivity feedback. This transient results in a rapid heat-up of the fuel, fuel melting, and pin failures, but near-nominal primary sodium conditions.

The selection of the PLOF+ and UTOP+ transient scenarios is not an indication of their specific importance to future SFR licensing efforts, or their likelihood or occurrence. The purpose of the inclusion of the two transients was to allow for the assessment of phenomena related to fuel pin failure with and without fuel melting and radionuclide transport within the primary sodium at near-nominal temperatures and at greatly-elevated temperatures.

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[1] The plus sign has been added to the transient name to indicate that the transient is more severe than the historical PLOF definition, which does not include degraded decay heat removal.

[2] The plus sign has been added to the transient name to indicate that the transient is more severe than the historical UTOP definition, which is typically of smaller magnitude and without degraded radial feedback.

Additional phenomena related to extremely unlikely metal fuel SFR transients, such as core disruption accidents (core energetics), were excluded from the trial MST calculation. It is assumed that SFR vendors will perform analyses that demonstrate the extremely low frequency of occurrence, or impossibility, of such events for their reactor design, therefore alleviating the vendor from the obligation to perform MST assessments for such reactor transients.

The transient scenarios were modeled using the SFR severe accident code SAS4A/SASSYS-1 [7]. Although SAS4A/SASSYS-1 does not include radionuclide tracking, the code does assess fuel pin failure and the associated fuel pin and primary system conditions. This information was used in subsequent MST analysis steps to determine the extent of radionuclide release from the failed fuel and behavior in the primary system. For the PLOF+ scenario, SAS4A/SASSYS-1 predicted the failure of all fuel pins within the core (48,780) due to the extended period of time at elevated temperature. However, as mentioned previously, no fuel melting is predicted. For the UTOP+, melting was conservatively assumed to occur in 10 assemblies (total of 2,710 fuel pins), as SAS4A/SASSYS-1 results were combined with experimental findings from the TREAT M-Series tests [8].

4.3.In-pin Radionuclide Distribution and Failed Fuel Radionuclide Release

During the irradiation of sodium-bonded metal fuel, radionuclides may migrate from the fuel matrix to the bond sodium and fission gas plenum, as shown in Figure 4. This phenomenon is important in determining the radionuclides available for release from the fuel pin following fuel pin failure. The findings of ANL-ART-38 were utilized to determine the extent of radionuclide migration for each of the three fuel batches of the reference reactor, each at different burnup levels.

The assessment of radionuclide release from failed metal fuel pins differed for the two transients assessed. For the PLOF+ transient, fuel failure occurred without fuel melting. In this scenario, shown in Figure 5, the contents of the fission gas plenum and bond sodium are assumed to be swept out of the fuel pin and may temporarily void the coolant channel, subsequently forming bubbles in the primary sodium. For the ULOF+ transient, the rapid rise in power causes in-pin fuel melting and relocation to occur before cladding failure. Once failure occurs, both molten fuel and radionuclides are ejected from the fuel pin, as shown in Figure 6. As there is no computer code available to model the release of radionuclide from failed metal fuel, a combination of findings from ANL-ART-38 and chemistry modeling using HSC [9] were utilized to estimate the extent of radionuclide release.

4.4.Radionuclide Release to the Cover Gas (Bubble Transport and Vaporization)

Following fuel pin failure, gases (mostly noble gases) released from the fuel pin will create bubbles within the primary sodium. Some of the non-gaseous radionuclides released from the fuel pin will become entrained within the bubbles as aerosols. As the bubble moves through the primary sodium, some of these aerosols will be removed from the bubble due to diffusion, deposition, sedimentation, and other phenomena. Also, some radionuclide gases/vapors, such as cesium, may condense as the bubble reaches colder regions of the primary sodium pool. If these radionuclides are not removed from the bubble, they will bypass the sodium pool and be released to the cover gas.

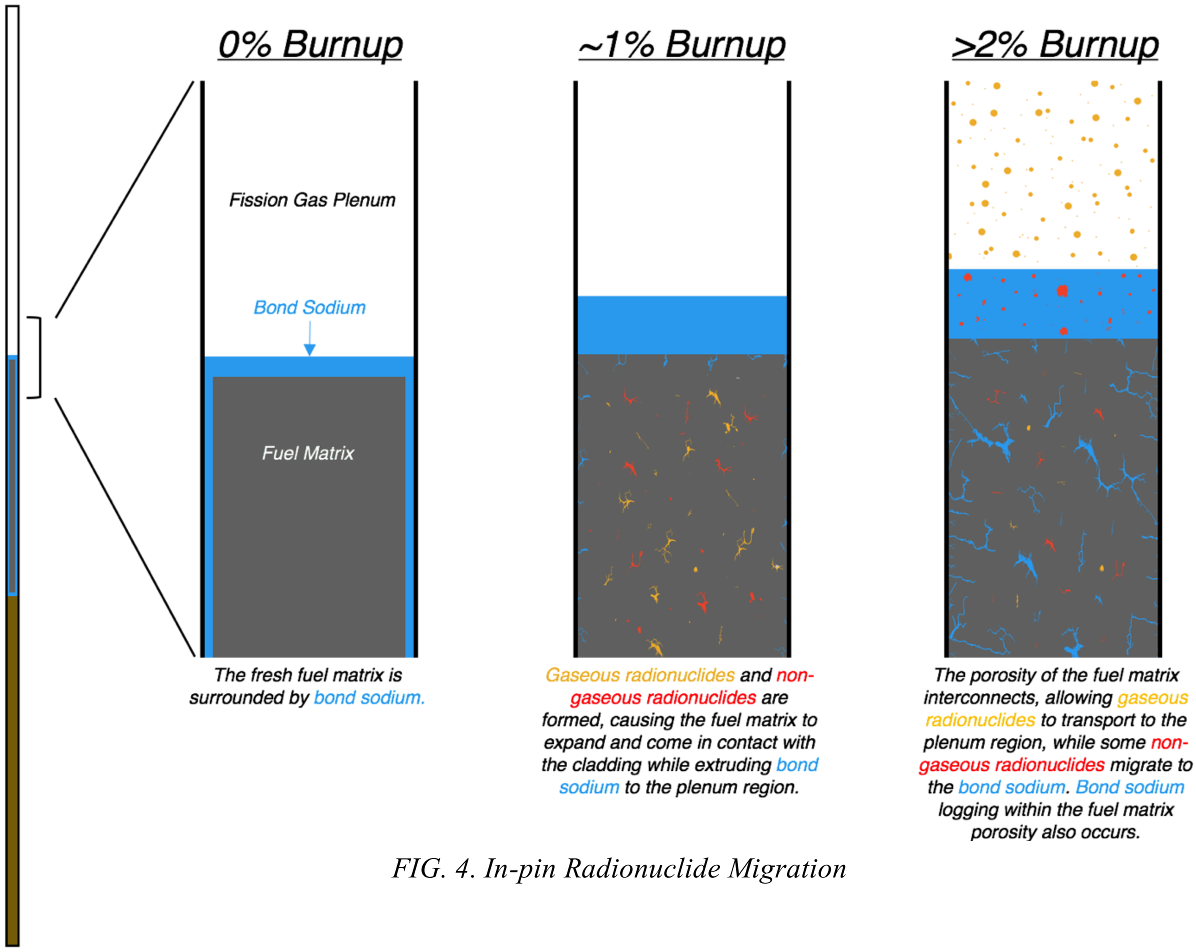

FIG. 4. In-pin Radionuclide Migration

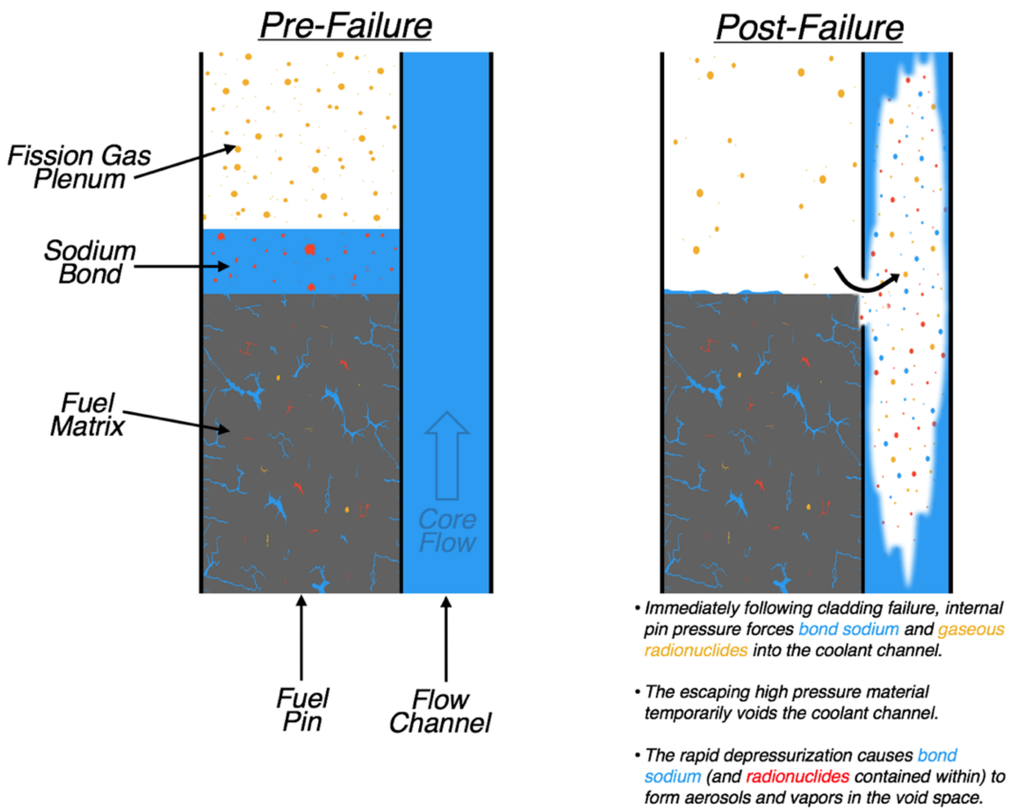

FIG. 5. Fuel Pin Failure Phenomena – No Melting

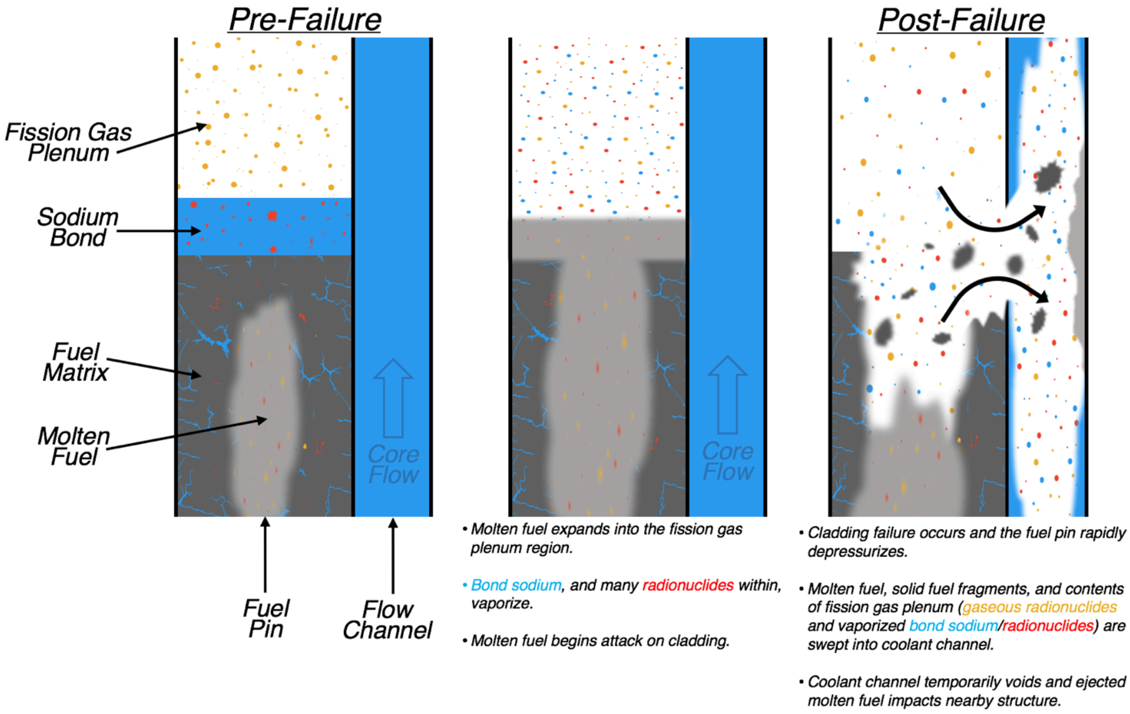

FIG. 6. Fuel Pin Failure Phenomena – With Melting

To assess these phenomena, an unnamed computer code developed during the Integral Fast Reactor (IFR) project was utilized. More detail on this analysis can be found in ref [10]. Although limited, the computer code predicted small radionuclide removal fractions (decontamination factors–DF of ~3) from the bubbles for the PLOF+ transient, as the primary sodium pool was at very high temperatures. For the UTOP+, radionuclide removal from the bubbles was high (DF's 60-150), since the primary sodium pool was at near-nominal conditions.

If radionuclides directly enter the primary sodium after fuel pin failure, or enter the primary sodium after removal from bubbles, it may be possible for the radionuclides to vaporize to the cover gas region. HSC was again used to explore this phenomenon utilizing a custom developed thermodynamic database. Due to large uncertainties regarding radionuclide entrainment within bubbles and removal from bubbles, bubble transported dominated radionuclide vaporization as the pathway for radionuclides to reach the cover gas region.

4.5.Cover Gas Region and Containment Analysis

Although the SFR-specific models of CONTAIN-LMR are currently being transferred to MELCOR for the assessment of SFR severe accidents [11], the newly developed code was not yet available for use during the current project. Attempts to utilize a developmental version of CONTAIN-LMR encountered several modeling issues[3]. Therefore, a simplified computer code was utilized to assess radionuclide behavior and release from the cover gas region and containment. The computer code utilized an exponential decay function for aerosol removal, with decay constants based on past experimentation, and also considered radionuclide decay and the daughter products of the noble gases. Assigned leakage rates were

[3] The CONTAIN-LMR modelling issues are described in ref [4]. Many of these issues have since been rectified and a revised CONTAIN-LMR analysis has been performed and is described in ref [12].

utilized to determine releases from the cover gas region to containment and from containment to the environment.

In general, noble gas release was dominant in terms of activity released to the environment. This is due to the fact that noble gases are not soluble in sodium and bypass the sodium pool, and also do not condense within the cover gas region or containment. Therefore, the only removal mechanism for noble gases released from the fuel is radioactive decay and hold-up in the cover gas region and containment. Complete radionuclide release results are available in ref [4].

4.6. Offsite Consequence Analysis

WinMACCS [13] was utilized for the assessment of offsite consequences, with a focus on the total effective dose equivalent (TEDE) since it is the basis of several U.S. NRC requirements [14]. The results for the UTOP+ scenario are shown in Figure 7, which illustrates that offsite consequences of the scenario were well below 25 rem TEDE even at distances very close to the containment building. The offsite doses were larger for the PLOF+ transient scenario, as the extent of fuel damage was greater than in the UTOP+, but were below the 25 rem TEDE level except the most severe uncertainty case at distances in very close proximity to the reactor building. Complete offsite dispersion results are available in ref [4].

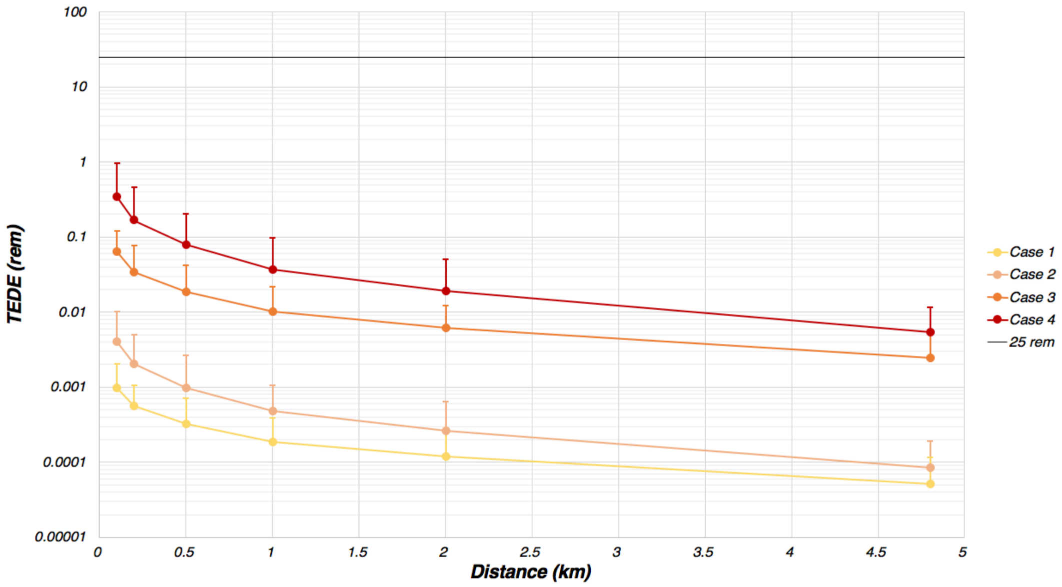

FIG. 7. UTOP+ Offsite Dispersion Results[4]

5. Sensitivity Calculation

The sensitivity calculation explored which radioisotopes and transport/retention phenomena were of most importance for offsite consequence (both dose and land contamination). Each radionuclide was assigned a dose-weighting and then tracked through the potential release pathways using simplified models. As the example shown in Figure 8 illustrates, dose-weighting the reactor inventory identified those radionuclides with the highest offsite consequence, along with which retention mechanisms were the most influential.

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[4] Four cases were examined for the UTOP+ transient scenario, with differing assumptions regarding radionuclide bubble behavior and leakage rates from the cover gas region. Details on each uncertainty case are provided in ref [4].

Sensitivity calculations were also performed for specific transport/retention phenomena. For example, separate calculations examined the impact of increasing or decreasing radionuclide retention within the fuel, radionuclide removal from bubbles, aerosol deposition, and cover gas region and containment leakage rates. The findings of this analysis, in terms of general importance to offsite consequence, are shown in Table II. The magnitude of radionuclide removal from bubbles within the primary sodium, and the associated uncertainties, proved to be the most influential phenomenon, followed by the radionuclide release fractions from failed fuel pins, especially for the actinides and lanthanides.

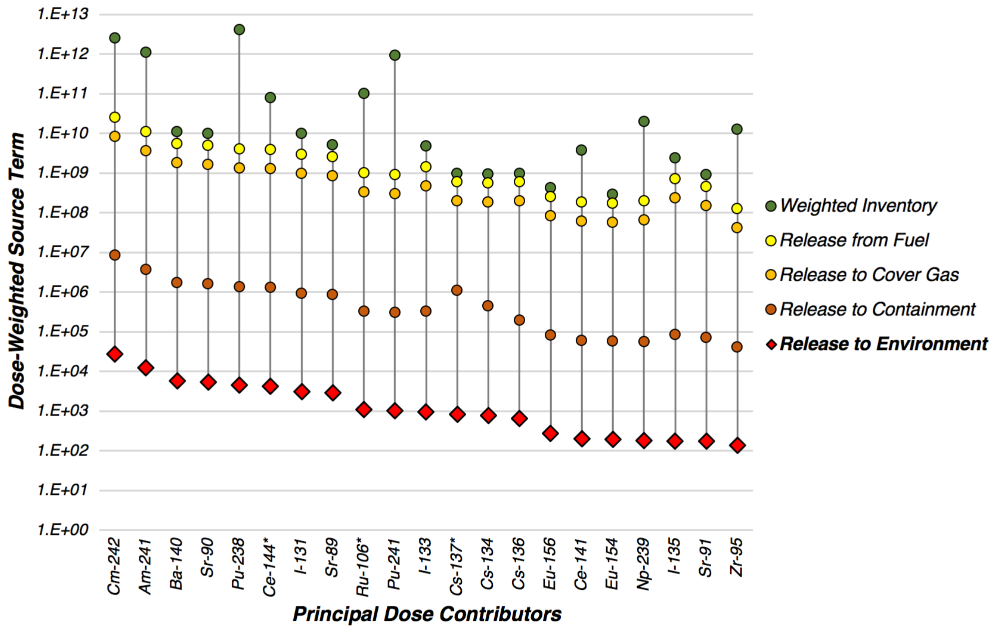

FIG. 8. Example Dose-Weighted Inventory Release

TABLE II: Sensitivity Analysis Transport/Retention Phenomena Findings

Phenomena	Importance
Pool Bypass (Bubble Transport)	Very High
Fuel Release Fractions (Actinide/Lanthanide)	High
Aerosol Deposition/Removal	Medium
Reactor Head/Containment Leak Rate	Medium
Pool Vaporization	Low
Noble Gas Decay Chains	Low

6. Conclusions and Future Research Prioritization

The main conclusion of the trial MST calculation project is that a SFR MST calculation is possible, with certain limitations, utilizing currently available tools and models. Gaps in models and data regarding some phenomena result in uncertainties or the use of conservative assumptions that could make the process of justifying reduced emergency planning zones or plant sites difficult for vendors. The gaps could also potentially impact design decisions that are dependent on source term analyses.

6.1.Gap Prioritization

Gaps in computational tools and data were reviewed and grouped into several categories based on the associated radionuclide transport/retention phenomena. These categories were then ranked by their importance to offsite consequence, based on the findings of the sensitivity analysis, and also the ability to resolve gaps using conservative assumptions in place of additional research. The final prioritization of the categories is shown in Table III. Bubble transport was determined to be the highest ranking gap group, as it provides a mechanism to bypass the sodium pool, a major radionuclide barrier, and has large uncertainties with little available data. This is followed by in-pin radionuclide migration and radionuclide release from failed fuel pins. Although conservative assumptions are easy to apply to this gap group, they can lead to unrealistic results in subsequent analysis steps.

TABLE III: Ranking of Gap Categories

Ranking	Gap Group	Notes
1	Bubble Transport	Represents a potential mechanism to bypass a major radionuclide barrier (the sodium pool). Very high importance in sensitivity calculations, and direct impact on non-noble gas radionuclide transport. Difficult to determine if analysis assumptions are realistic, and significant impact on subsequent analysis steps.
2	In-Pin Migration and Release	Determines initial radionuclide release fractions. High importance in sensitivity calculations. Conservative assumptions are straightforward to apply, but assumptions propagate through subsequent analysis steps, resulting in potentially unrealistic releases of transuranics.
3	Aerosol Behaviour	Aerosol deposition/condensation is a significant retention mechanism with medium importance in sensitivity calculations. Data regarding deposition, condensation, and chemical interactions are not as well-established as LWR data. Codes, such as MELCOR, have ability to model phenomena, but data are a necessary input.
4	Hold-up/Leakage	The delay in radionuclide releases due to hold-up in the cover gas region and containment. Medium importance in sensitivity calculations. Assumptions are straightforward to apply and characterize, as shown in trial MST calculation where multiple leakage values were assumed.
5	Vaporization	Vaporization of radionuclides from the sodium pool. Many gaps in the modelling of phenomena, but was low importance in sensitivity calculations (and in trial MST calculation). However, the relative importance of vaporization could increase, if bubble transport calculations can be shown to be overly conservative.
6	Dispersion	Radionuclide characteristics and dose conversion factors. Although these factors have a direct impact on offsite consequence, the magnitude of their influence is likely lower than the factors related to radionuclide release fractions from the plant.

6.2. Research Recommendations

In conjunction with the gap analysis, a series of research recommendations were formulated, as shown in Table IV. Experimentation regarding radionuclide release from failed fuel (including fuel pin depressurization) and bubble transport through liquid sodium are the highest ranking recommendations, along with the formal completion of the IFR bubble transport code described in Section 2.4. Additional metal fuel post-irradiation examination (PIE) and melt tests of high burnup metal fuel in liquid sodium are seen as the major research areas for in-pin radionuclide migration and failed fuel pin radionuclide release. For aerosol behavior, completion of the SFR version of MELCOR is the most important research area. The research areas within the other gap groups were found to be of lesser importance.

7. Acknowledgements

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TABLE IV: Research Recommendations

Group	Recommendations
Bubble Transport	• Formal completion of the IFR bubble code, including development of documentation and code licensing pathway. • Experimentation regarding failed fuel pin blowdown and entrainment of released radionuclides in bubble (cross-cutting research with reactivity effects of channel voiding and structural impacts of blowdown on neighbouring fuel pins). • Experimentation regarding removal of radionuclides from bubble traveling through sodium pool.
In-Pin Migration and Release	• Continued metal fuel PIE to determine radionuclide migration within fuel pin during irradiation. • Experimentation regarding radionuclide release from failed high burnup fuel pins at high temperatures (above fuel melting point) in liquid sodium.
Aerosol Behaviour	• Continued development of SFR version of MELCOR. • Assessment of available data regarding deposition/condensation/chemical interactions within cover gas region and containment.
Hold-up/Leakage	• Response of reactor head seals during transient conditions.
Vaporization	• Investigation of non-homogeneous mixing of radionuclides in liquid sodium.
Dispersion	• Assessment of applicability of dose conversion factors and deposition assumptions to chemical and physical radionuclide forms likely in SFR radionuclide release.

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