August 11, 2023

TP-LIC-LET-0093
Project Number 99902100

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
ATTN: Document Control Desk

Subject: Transmittal of TerraPower, LLC Topical Report, "Radiological Source Term Methodology Report," Revision 0

This letter transmits the TerraPower, LLC (TerraPower) Topical Report "Radiological Source Term Methodology Report," Revision 0 (enclosed). The report contains an overview and description of the model developed to evaluate mechanistic source terms for the Natrium™ Plant[1].

TerraPower requests the NRC's review and approval of the source term evaluation model presented in this report for use by future applications utilizing the Natrium design.

TerraPower requests that a nominal review duration of 12 months be considered.

The report contains proprietary information and as such, it is requested that Enclosure 3 be withheld from public disclosure in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." An affidavit certifying the basis for the request to withhold Enclosure 3 from public disclosure is included as Enclosure 1. Enclosure 3 also contains ECI which can be disclosed to Foreign Nationals only in accordance with the requirements of 15 CFR 730 and 10 CFR 810, as applicable. Proprietary and ECI materials have been redacted from the report provided in Enclosure 2; redacted information is identified using [[ ]]$^{(a)(4)}$, [[ ]]^{ECI}, or [[ ]]$^{(a)(4), ECI}$.

This letter and enclosures make no new or revised regulatory commitments.

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[1] Natrium is a TerraPower and GE-Hitachi technology.

If you have any questions regarding this submittal, please contact Ryan Sprengel at rsprengel@terrapower.com or (425) 324-2888.

Sincerely,

Ryan Sprengel
Director of Licensing, Natrium
TerraPower, LLC

Enclosure:
1. TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure (10 CFR 2.390(a)(4))
2. TerraPower, LLC Topical Report, "Radiological Source Term Methodology Report," Revision 0 – Non-Proprietary (Public)
3. TerraPower, LLC Topical Report, "Radiological Source Term Methodology Report," Revision 0 – Proprietary (Non-Public)

cc: Mallecia Sutton, NRC
William Jessup, NRC
Nathan Howard, DOE
Jeff Ciocco, DOE

ENCLOSURE 1

TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure (10 CFR 2.390(a)(4))

Enclosure 1
TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure
(10 CFR 2.390(a)(4))

I, George Wilson, hereby state:

1. I am the Vice President, Regulatory Affairs and I have been authorized by TerraPower, LLC (TerraPower) to review information sought to be withheld from public disclosure in connection with the development, testing, licensing, and deployment of the Natrium™ reactor and its associated fuel, structures, systems, and components, and to apply for its withholding from public disclosure on behalf of TerraPower.
2. The information sought to be withheld, in its entirety, is contained in Enclosure 3, which accompanies this Affidavit.
3. I am making this request for withholding, and executing this Affidavit as required by 10 CFR 2.390(b)(1).
4. I have personal knowledge of the criteria and procedures utilized by TerraPower in designating information as a trade secret, privileged, or as confidential commercial or financial information that would be protected from public disclosure under 10 CFR 2.390(a)(4).
5. The information contained in Enclosure 3 accompanying this Affidavit contains non-public details of the TerraPower regulatory and developmental strategies intended to support NRC staff review.
6. Pursuant to 10 CFR 2.390(b)(4), the following is furnished for consideration by the Commission in determining whether the information in Enclosure 3 should be withheld:
    a. The information has been held in confidence by TerraPower.
    b. The information is of a type customarily held in confidence by TerraPower and not customarily disclosed to the public. TerraPower has a rational basis for determining the types of information that it customarily holds in confidence and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application and substance of that system constitute TerraPower policy and provide the rational basis required.
    c. The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR 2.390, it is received in confidence by the Commission.
    d. This information is not available in public sources.
    e. TerraPower asserts that public disclosure of this non-public information is likely to cause substantial harm to the competitive position of TerraPower, because it would enhance the ability of competitors to provide similar products and services by reducing their expenditure of resources using similar project methods, equipment, testing approach, contractors, or licensing approaches.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on: August 11, 2023

George Wilson

George Wilson
Vice President, Regulatory Affairs
TerraPower, LLC

ENCLOSURE 2

TerraPower, LLC Topical Report
"Radiological Source Term Methodology Report" Revision 0

Non-Proprietary (Public)

Controlled Document - Verify Current Revision

Topical Report

Document Number:	TP-LIC-RPT-0003	**Revision:**	0
Document Title:	Radiological Source Term Methodology Report		
Functional Area:	Licensing	**Engineering Discipline:**	N/A
Effective Date:	8/11/2023	**Released Date:**	8/11/2023
		Page:	1 of 106

Approval

Title	Name	Signature	Date
Originator, Licensing Engineer	Matthew Presson	Electronically Signed in Agile	8/11/2023
Reviewer, Licensing Manager	Nick Kellenberger	Electronically Signed in Agile	8/11/2023
Approver, Director of Licensing	Ryan Sprengel	Electronically Signed in Agile	8/11/2023

Export Controlled Content: **Yes** ☐ **No** ☒

REVISION HISTORY

Revision No.	Effective Date	Affected Section(s)	Description of Change(s)
0	8/11/2023	All	Initial Release

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EXECUTIVE SUMMARY

This report documents the radiological source term evaluation model (EM) development process for the Natrium™ reactor, a TerraPower & GE-Hitachi Technology. The resulting EM, and items identified which require further development, are described. The report contains eight chapters as well as an appendix containing sample calculations.

Chapter 1 discusses the overall objective and scope of the report, the regulatory requirements and guidance used in the EM development process, a high-level description of the Natrium design, and identifies the safety systems and design basis accidents (DBAs) that pertain to the Source Term EM development and how the DBAs fit within the overall identification of event types addressed.

Chapter 2 discusses the EM capability requirements development. A four-step process was undertaken to define the capabilities of the Source Term EM. These steps included:

1. Specify analysis purpose, transient class, and power plant class (Section 2.1)
2. Specify figures of merit (FOMs) (Section 2.2)
3. Identify systems, components, phases, geometries, fields, and processes that must be modeled (Section 2.3)
4. Identify list of important key phenomena (Section 2.4)

Chapter 3 discusses development of the EM assessment base and is generally focused on addressing applicable aspects of Element 2 of Regulatory Guide (RG) 1.203, i.e., Evaluation Model Development and Assessment Process (EMDAP). This chapter includes discussion of the assessment base objectives, scaling analysis and similarity criteria, existing data needed to complete the EM validation database, evaluation of integral effects test (IET) distortions and separate effects test (SET) scaleup capability, and experimental uncertainties determination.

Chapter 4 discusses EM development including the associated plan, a listing of computer codes considered for inclusion in the EM, computer codes upstream of the EM, code selection gaps, the EM structure, and the strategy for DBA modeling.

Chapter 4 further discusses the conservative methods for EM applications from three perspectives:

1. In contrast with best-estimate-plus uncertainty methods.
2. The required justifications for adopting the conservative methods.
3. With respect to cases where code coupling is required.

In general, conservative methods are developed for primary boundary and initial conditions, e.g., plant initial conditions, core power distribution, and other characteristics of the fuel operational and safety systems, thermal hydraulics, etc. The events that will be specifically considered fall across normal operations, system leakage scenarios, selected licensing basis events (LBEs), DBAs, fuel handling accidents (FHAs), and dose mapping for environmental qualification evaluations.

Chapter 5 discusses the EM adequacy assessment. The evaluation methodology focuses on radionuclide releases from normal operation as well as those from anticipated fuel defects and neutron activation. The source term methodology includes evaluation of source term for:

- Effluents
- Radwaste system design

- Shielding design
- Equipment qualification (EQ)

The source terms considered span normal operations, system leakage scenarios, plausible accident scenarios and emergency zone planning. The Natrium methodology is compared to the expectations noted in RG 1.183, Regulatory Position (RP) 2, i.e., attributes of an acceptable alternative source term; RG 1.183 RP2 has been determined to be relevant from the following perspectives:

- A source term must be based upon major accidents for purposes of design analyses or consideration of possible potential accidental events.
- The source term must be expressed in terms of times and rates of appearance of radioactive species released and the chemical forms of iodine release.
- The source term must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events.
- A source term must have a defensible technical basis supported by sufficient experimental and empirical data, be verified, and validated.
- A source term must be peer-reviewed by appropriately qualified subject matter experts.

Consequently, an initial list comprised of 19 potential sources was constructed to serve as the basis for evaluation and determining the adequacy of the Source Term EM (where the EM is comprised of the analysis codes: [[ ]](a)(4), RADTRAD, and [[ ]](a)(4)) via EMDAP of RG 1.203. The final step of RG 1.203 (Step 20), i.e., the determination of EM biases and uncertainties, will address the prediction of the FOMs through incorporation of biases and uncertainties into the various code mathematical models considering: (i) characterization of the sources of uncertainty, (ii) the quantification of the propagation of uncertainties through the various codes and, (iii) consideration of sensitivity analyses of the calculational outputs. Each code within the Source Term EM will be evaluated independently to characterize the sources of uncertainties as well as the propagation of uncertainties. Also, the inter-relationships of the codes relative to one another will be evaluated relative to the propagation of uncertainties. The uncertainty characterization will identify the various ingredients as epistemic or aleatory where each type will be treated appropriately through consideration of either probability density functions or cumulative distribution functions. Finally, a similar approach will be performed to define the uncertainty treatment of high-risk phenomena.

Chapter 6 discusses sample analyses that will be performed to demonstrate the methodology. The calculations will demonstrate how the various codes (components) of the EM will be used in conjunction with one another.

Chapter 7 discusses the overarching EM adequacy decision.

Chapter 8 describes the limitations of this radiological Source Term EM and identifies five explicit items related to limitations of the EM. These limitations center on the reactor design, fuel design, sodium bonding, sodium pool scrubbing, and the bounds of the model.

ACRONYMS

Acronym	Definition
AHX	Sodium-Air Heat Exchanger
ALARA	As Low As Reasonably Achievable
ALI	Annual Limits on Intake
AOO	Anticipated Operational Occurrence
AST	Alternate Source Term
BDBE	Beyond Design Basis Event
BLTC	Bottom Loading Transfer Cask
CFR	Code of Federal Regulations
DAC	Derived Air Concentration
DBA	Design Basis Accident
DBE	Design Basis Event
DCH	Decay Heat Package
DID	Defense-in-Depth
DOE	Department of Energy
EAB	Exclusion Area Boundary
EBR	Experimental Breeder Reactor
EM	Evaluation Model
EMDAP	Evaluation Model Development and Assessment Process
EPZ	Emergency Planning Zone
EQ	Equipment Qualification
ESS	Energy Island Salt Heat Transport System
EVHM	Ex-Vessel Handling Machine
EVST	Ex-Vessel Storage Tank
FAB	Fuel Auxiliary Building
F-C	Frequency-Consequence
FFTF	Fast Flux Test Facility
FHA	Fuel Handling Accident
FHB	Fuel Handling Building
FOM	Figure of Merit
FSAR	Final Safety Analysis Report
GEH	GE-Hitachi
GV	Guard Vessel
IAC	Intermediate Air Cooling
IAEA	International Atomic Energy Association
IET	Integral Effects Test
IHT	Intermediate Heat Transport System
IHX	Intermediate Heat Exchanger
ISP	Intermediate Sodium Pump
IVS	In-Vessel Storage

Acronym	Definition
IVTM	In-Vessel Transfer Machine
LBE	Licensing Basis Event
LMP	Licensing Modernization Project
LOCA	Loss of Coolant Accident
LPZ	Low Population Zone
LWR	Light Water Reactor
MHA	Maximum Hypothetical Accident
MST	Mechanistic Source Term
Na	Sodium
NAC	Sodium Chemistry Package
NSRST	Non-Safety-Related with Special Treatment
NSS	Nuclear Island Salt System
NST	Non-Safety-Related with No Special Treatment
OQE	Other Quantified Events
PHT	Primary Heat Transport System
PIC	Pool Immersion Cell
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PRISM	Power Reactor Innovative Small Module
PSP	Primary Sodium Pump
RAC	Reactor Vessel Air Cooling
RES	Reactor Enclosure System
RG	Regulatory Guide
RI	Reactor Internals
RF	Release Fraction
RN	Radionuclide
RP	Regulatory Position
RSA	Reactor Support Assembly
RSF	Required Safety Function
RV	Reactor Vessel
RVH	Reactor Vessel Head
RWG	Gaseous Radioactive Waste System
RWL	Liquid Radioactive Waste System
RWS	Solid Radioactive Waste System
RXB	Reactor Building
SCG	Sodium Cover Gas System
SET	Separate Effects Test
SFR	Sodium-Cooled Fast Reactor
SHX	Sodium-Salt Heat Exchanger
SNL	Sandia National Laboratories
SPS	Sodium Processing System

Acronym	Definition
SR	Safety-Related
SRM	Staff Requirement Memorandum
SSC	Structure, System, and Component
TEDE	Total Effective Dose Equivalent
TES	Thermal Energy Storage System
TH	Thermal Hydraulic
TWR	Travelling Wave Reactor®
UFP	Used Fuel Pool
ULOF	Unprotected Loss of Flow
ULOF+	Unprotected Loss of Flow with Degraded Pump Coastdown

1 INTRODUCTION

This report addresses the Natrium radiological Source Term EM development process, the resulting EM, and identifies items which require further development. Overarching TerraPower methodology development guidance coupled with the U.S. Nuclear Regulatory Commission (NRC) guidance (RG 1.203, *Transient and Accident Analysis Methods*) [1] were used to guide the Source Term EM development process. As noted throughout this report, not all aspects of RG 1.203 are directly applicable to the Source Term EM development process. Nonetheless, the overall Source Term EM development process generally adheres to the RG 1.203 process. The adequacy of the Source Term EM was achieved by following, where appropriate, the RG 1.203 EMDAP which is shown in flow chart form in RG 1.203, Figure 1. Note that EMDAP consists of four elements followed by an “Adequacy Decision” when the contents of the four elements are completed:

Element 1	Establish requirements for EM capability—see Chapters 1 and 2
Element 2	Develop assessment base—see Chapter 3
Element 3	Develop EM—see Chapter 4
Element 4	Assess EM adequacy—see Chapters 5 and 7

Element 1 focuses on establishing the boundary conditions for determining: (i) the necessary capabilities of the EM by identifying the physics that should be contained in the EM, (ii) the geometries of the subject nuclear system that must be evaluated with the EM, (iii) the safety margin of the subject nuclear system using key measurable physical parameters that are closely associated with the plant operational and accident limits — commonly labeled “figures-of-merit”, and (iv) the adequacy of the EM that is to be developed in Element 3. Element 1 consists of the first four steps of EMDAP.

Element 2 encompasses the effort required to adequately assemble experimental data for use as reference for determining the adequacy of the EM. The data captured in Element 2 must be relatable to the full-sized nuclear system using a hierarchical scaling law approach that contains a way to measure the geometrical correspondence, physical properties, representative events, representative sequences of events, and transient timing of events with respect to the full-sized nuclear system. Element 2 consists of Steps 5 through 9 of EMDAP.

Element 3 includes the activities of (i) establishing an EM development plan and (ii) constructing the EM. The action of creating an EM development plan (identified as Step 10 in EMDAP) is the key activity of EMDAP. An EM development plan includes the following ingredients (see RG 1.203, Appendix B, pp. B-9 to B-10): (a) the software quality assurance plan, (b) the software requirements specification, (c) documentation of the software design and implementation, (d) the source code verification test report, (e) the validation testing report, and (f) the installation package plus program upgrade documentation. Within these sections and associated documentation rest the description of phenomena that must be contained within the EM, the ways and means for demonstrating closure for both code verification and solution verification of the EM, and the measures that are to be used to determine whether or not the EM is capable of calculating all key phenomena within the nuclear reactor components and within the system as a whole for all transients listed in Element 1. This guides the specification of experiments and required measurement uncertainties, the acceptable distortion levels of experiments to be used to generate validation data, the scale-up of experimental data recorded in experimental facilities smaller than the full-sized plant, the validation metrics, and the limits within which the determination of EM adequacy will be made. In a sense, all activities in both Elements 1 and 2 are inputs to the EM development plan, and the remainder of Element 3 and all of Element 4 are steps that guide the execution of the EM development plan. Element 3 consists of Steps 10 through 12 of EMDAP.

Element 4 describes the performance of the EM development plan via (i) bottom-up considerations, i.e., model pedigree, performance of calculations to enable validation studies to be performed through model scalability, and (ii) via top-down considerations, i.e., the demonstration of scalability of integrated calculations for the transient class under consideration. Element 4 consists of Steps 13 through 20 of EMDAP.

EM Adequacy Decision, the final step in EMDAP, is performed by comparing the results obtained throughout EMDAP to the measures of success prescribed in the EM development plan (Step 10 within Element 3). Successful completion of the EM development plan, as demonstrated by meeting all of the requirements of the EM development plan, enables the required plant event analyses to be performed for licensing purposes.

Certain aspects of RG 1.203 EMDAP do not lend themselves to the Source Term EM development process. Consistent with TerraPower EM development guidance some aspects of EMDAP are correspondingly not part of the Source Term EM development process as noted throughout this report.

1.1 Objective and Scope

The Source Term EM is used to evaluate the radiological consequences of quantified events (see Figure 1-1). This report documents the development of the Natrium Source Term EM. The report is organized into eight chapters.

Chapter 1 provides the introduction to the process followed for the Source Term EM and includes discussion of regulatory requirements and guidance, high level descriptions, considerations for classifying Natrium Safety Systems, and the types of events where the Source Term EM is applicable.

Chapter 2 provides the Source Term EM capability requirements and includes discussion of the four-step process followed to define the Source Term EM required capabilities. The four steps include:

1) Specifying the analysis purpose, transient class, and power plant class
2) Specifying FOMs
3) Identifying systems, components, phases, geometries, fields, and processes that must be modeled
4) Listing important phenomena

Chapter 3 provides the EM assessment base development including the objectives, scaling analysis and similarity criteria, identification of existing data needed to complete the EM validation database, evaluation of IET distortions and SET scaleup capability, and experimental uncertainties evaluation.

Chapter 4 provides the EM development including the plan, EM structure, and closure models. This chapter includes discussion of the codes [[ ]](a)(4), and RADTRAD.

It also provides discussion on conservatisms used when applying the EM, and includes general conservatisms and event-specific conservatisms.

Chapter 5 provides the EM adequacy assessment including the closure relations (bottoms-up), integrated EM (top-down), and EM biases and uncertainties.

Chapter 6 is reserved for EM sample analysis results once they are available.

Chapter 7 provides the overall EM adequacy decision.

Chapter 8 describes the limitations of this radiological Source Term EM and identifies five explicit items related to the EM limitations.

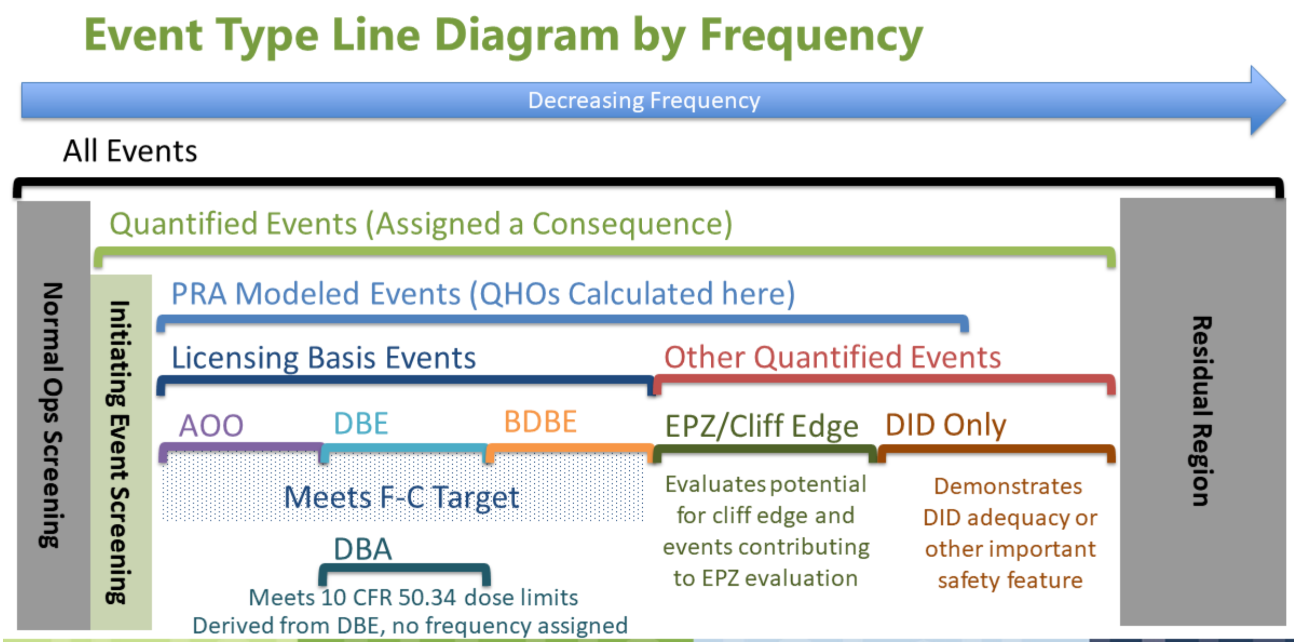

Figure 1-1. Event Type Line Diagram by Frequency

1.2 Regulatory Requirements and Guidance

The NRC defines source term specifically in 10 CFR 50.2 as,

"The magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form and the timing of their release."

The term inventory typically refers to the radionuclides (RNs) contained in a plant system, fuel assembly, core, etc. For the purposes of this document, the term source is used in a broader sense due to the range of radiological evaluations that will be affected by the Source Term EM. In one evaluation, the inventory in a system is not a source until it reaches a release point. In a separate evaluation, the inventory in system piping or the fuel assembly is the source for a subsequent radiological evaluation. Therefore, in some portions of the discussion in this document the two terms may be applied interchangeably. This is done while acknowledging that, for individual subsequent source term evaluations that follow this plan, a distinction between inventory and actual source term may need to be clear to avoid confusion.

The source term methodology defines how to determine the in-building radiological source terms that account for radioactive material composition and activity, as well as the chemical and physical properties of the material within the building that are available for potential release to the environment. In other words, the source term includes not only the RN inventory of the fuel assemblies but also the release timing and the rate of release of the RNs from the core to the containment, as well as the effect of RN removal mechanisms from the containment.

A Maximum Credible Accident is an accident postulated that would result in a potential hazard that would not be exceeded by any other accident considered credible during the lifetime of a facility. Historically, such an accident is known as a DBA. DBAs are intended to be surrogates to enable deterministic evaluation of the response of engineered safety features. These accident analyses are intentionally conservative to compensate for known uncertainties in accident progression, RN transport, and atmospheric dispersion.

The Source Term EM development process considers U.S. NRC guidance on the EMDAP as established in RG 1.203. [1] Note that while adopting the licensing modernization project (LMP) framework [2], endorsed as RG 1.233 [3], the Source Term EM development process does not meet verbatim conformance with RG 1.203 but rather considers the EMDAP as an industry best practice in methods development.

RG 1.183 [4], Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, outlines an acceptable alternate source term (AST). It notes that NRC staff does not expect to approve any source term that is not of the same level of quality as the source terms in NUREG-1465. To be considered acceptable, RG 1.183 asserts in Positions 2.1 through 2.5 that an AST must have the following attributes:

1. The AST must be based on major accidents, hypothesized for the purposes of design analyses or consideration of possible accidental events, that could result in hazards not exceeded by those from other accidents considered credible. The AST must address events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.
2. The AST must be expressed in terms of times and rates of appearance of radioactive fission products released into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.
3. The AST must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events. Risk insights may be used, not to select a single risk-significant accident, but rather to establish the range of events to be considered. Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.
4. The AST must have a defensible technical basis supported by sufficient experimental and empirical data, be verified, and validated, and be documented in a scrutable form that facilitates public review and discourse.
5. The AST must be peer-reviewed by appropriately qualified subject matter experts. The peer-review comments and their resolution will be part of the documentation supporting the AST.

Source term methods development does not have specific compliance requirements. As discussed above, the Source Term EM development considers guidance in RG 1.203 and will seek to satisfy

RG 1.183 Positions 2.1 through 2.5. However, the intended scope would be different from a typical system analysis EM development process. As existing regulations are tuned toward light water reactor (LWR) designs, many aspects are not directly applicable to Natrium design features such as metallic fuel and sodium coolant. A scenario-specific source term, also known as a mechanistic source term (MST), has been proposed for advanced LWRs and non-LWRs. The NRC has approved this approach and further issued draft review guidance on the pre-application engagement of advanced reactors. Examples of this process include SECY-93-092 [5], staff requirement memorandum (SRM) on SECY-93-092 [6], SECY-03-0047 [7], SRM on SECY-03-0047 [8], SECY-05-0006 [9], and SECY-16-0012 [10]. Moreover, the ASME/ANS Advanced Non-LWR Probabilistic Risk Assessment (PRA) Standard (ASME-RA-S-1.4 2021) [11] includes "Mechanistic Source Term Analysis" as one of the PRA elements. The high-level requirements for this element are:

1. The definition and characterization of release categories shall be sufficient for the requirements of the MST analysis and radiological consequence analysis.
2. The MST analysis shall assess the RN transport barriers and transport mechanisms for each release.
3. The MST and associated RN transport phenomena shall be calculated.
4. Uncertainties in the MSTs and associated RN transport phenomena shall be identified, characterized, and quantified to the extent practical. Key sources of model uncertainty and assumptions shall be identified, and their potential impact on the results shall be understood. Those sources of uncertainty that are not quantified shall be assessed via sensitivity evaluation(s).
5. The documentation of the MST analysis shall provide traceability of the work.

As ORNL/TM-2020/1719 [12] states,

"However, overly simplistic bounding accidents can be limited in their ability to assess weaknesses in the safety functions that could make an off-site radiological release event more likely. This dimension of reactor safety assesses how a design copes with challenges presented by the full spectrum of accident scenarios that could occur. How a design interacts with a range of different accident scenarios can be different from its response to particular bounding accident scenarios."

The accident source term will be based on a spectrum of credible severe accident events that include a substantial meltdown of the core with the subsequent release of appreciable quantities of RNs. Such an accident is known as the maximum hypothetical accident (MHA). Even if the postulated maximum credible DBA would occur, the resulting radiological consequences would be lower than those from the MHA.

The regulatory guidance on LMP [2] has been used in selecting events for the Natrium design. This risk-informed process is based on realistic assessments of plant performance leading to a set of LBEs. It is allowable in the LMP process for fuel failure to occur in LBEs but based on the performance characteristics of the design, it is expected that there will be no damage beyond local faults. When applying the defense-in-depth (DID) process, there will be sensitivity studies applied to additional events that may be just below the Beyond Design Basis Event (BDBE) frequency cutoff which could also have significant fuel failure to ensure there are no missed cliff edge effects due to event screening. These rare, but quantified, events take the place of the ad hoc, presumably bounding, hypothetical accidents. Any event proposed and analyzed will have an associated event sequence to maintain a mechanistic event progression.

Although the LMP is intended to have a flexible, performance-based approach for establishing scenario-specific licensing source terms, it puts the burden on the applicant to develop the technical basis (including experimental data) to support its proposed source terms. The use of scenario-specific source terms is based on sufficient understanding and assurance of plant and fuel performance, which are generally lacking in non-LWR applications.

The source term methodology also covers RN releases during normal operation. Anticipated fuel defect and neutron activation will result in some minor periodic release of RNs to the environment. Hence, the source term methodology also includes radiological source terms for effluents, radwaste system design, shielding design, and EQ.

Like standard LWR designs, the Natrium plant is subject to the regulations outlined in the *Code of Federal Regulations* (CFRs), with exceptions as requested and justified based on the design. The MSTs must at least address requirements outlined in 10 CFR Parts 20, 50 and 100. 10 CFR Part 20 is "STANDARDS FOR PROTECTION AGAINST RADIATION" and lists dose limits for workers and the public from plant operation. 10 CFR Part 50 is "DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES," lists requirements to license nuclear facilities. 10 CFR Part 100 is "REACTOR SITE CRITERA" and lists requirements for locating nuclear facilities.

Key sections of 10 CFR Parts 20, 50, and 100 that are related to the source term development process are:

- 10 CFR 20.1201 occupational dose limits for adults
- 10 CFR 20.1301 dose limits for individual members of the public
- 10 CFR 20.1302 compliance with dose limits for individual members of the public
- 10 CFR 20 Appendix B Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage
- 10 CFR 50.2 Definitions
- 10 CFR 50.34(b)(3) requirement for the final safety analysis report (FSAR) to define the kinds and quantities of radioactive materials expected to be produced in the operation of the facility and means to keep within the 10 CFR Part 20 limits
- 10 CFR 50.49 environmental qualification of electric equipment important to safety for nuclear power plants
- 10 CFR 100.11 determination of exclusion area, low population zone (LPZ), and population center distance for facility siting (note that this regulation is not directly applicable, but the code is still consistent with the criteria set forth in § 50.34(a)(1))
- Control Room Habitability (GDC-19 of Appendix A to 10 CFR Part 50 as well as ARDC-19 of RG 1.232)

Using this list of requirements, the following parameters have been collected and are further elaborated upon in subsequent sections of this report.

Normal Operation

- The effluents released on an ongoing basis due to normal operations

- Spent fuel source terms
- Primary coolant due to normal fuel defect
- Activation products generated from core neutron/gamma flux (e.g., tritium, sodium activation, etc.)
- Buildup of RNs in decontamination systems and associated solid, liquid, and gaseous wastes
- Personnel operations scenarios/ALARA

System leakage scenarios, includes consideration of the following:

- Cover gas cleanup system leak
- Sodium cleanup system leak
- Intermediate Heat Transport system (IHT) leak
- Gaseous Radioactive Waste system (RWG) system leak

LBEs - Plausible Accident Scenarios, includes consideration of the following:

- potential scenarios
- fuel failure fractions
- isotopic release fractions
- release pathways

FHA Scenarios

- FHA in-vessel
- FHA in Ex-Vessel Storage Tank (EVST)
- FHA in Spent Fuel Pool
- FHA in Washing Station
- FHA during Fuel Transfer

Plume exposure pathway emergency planning zone (EPZ) sizing Methodology

Neutronics Methodologies (including RN inventory generation)

Dose Mapping of Nuclear Facility for Subsequent EQ Evaluations

1.3 Plant Description

The Natrium reactor is a sodium-cooled fast reactor (SFR) that uses a fuel design and an operating environment that are significantly different from LWRs currently utilized in the United States. The Natrium reactor is an innovative design that facilitates rapid construction and achieves cost competitiveness and flexible operations through the adoption of new technology and a reimagined plant layout. Many of these advances are enabled through inherent safety features of pool-type SFRs with metal fuel. The design is based on early reactor technology developed in the US by the Department of Energy (DOE) and was developed from decades of research, design, and development from GE-Hitachi's (GEH) Power Reactor Innovative Small Module (PRISM) technology and TerraPower's Traveling Wave Reactor (TWR®) technology.

The general plant layout is shown in Figure 1-2 and is made up of two basic areas; a Nuclear Island where the reactor and associated support facilities reside and an Energy Island where thermal storage tanks and turbine facilities for generating electricity reside. Safety functions are made integral to the Reactor Vessel (RV) and support equipment is moved to separate structures in the Energy Island, resulting in a simplified Reactor Building (RXB). The design leverages the legacy of 40 reactor-years of EBR-II and FFTF operation. These two predecessor reactors demonstrated how SFRs can passively accommodate severe transients. The design capitalizes on the proven metal fueled SFR safety characteristics to minimize the number of safety-related (SR) structures, systems, and components (SSCs) needed to achieve safety goals.

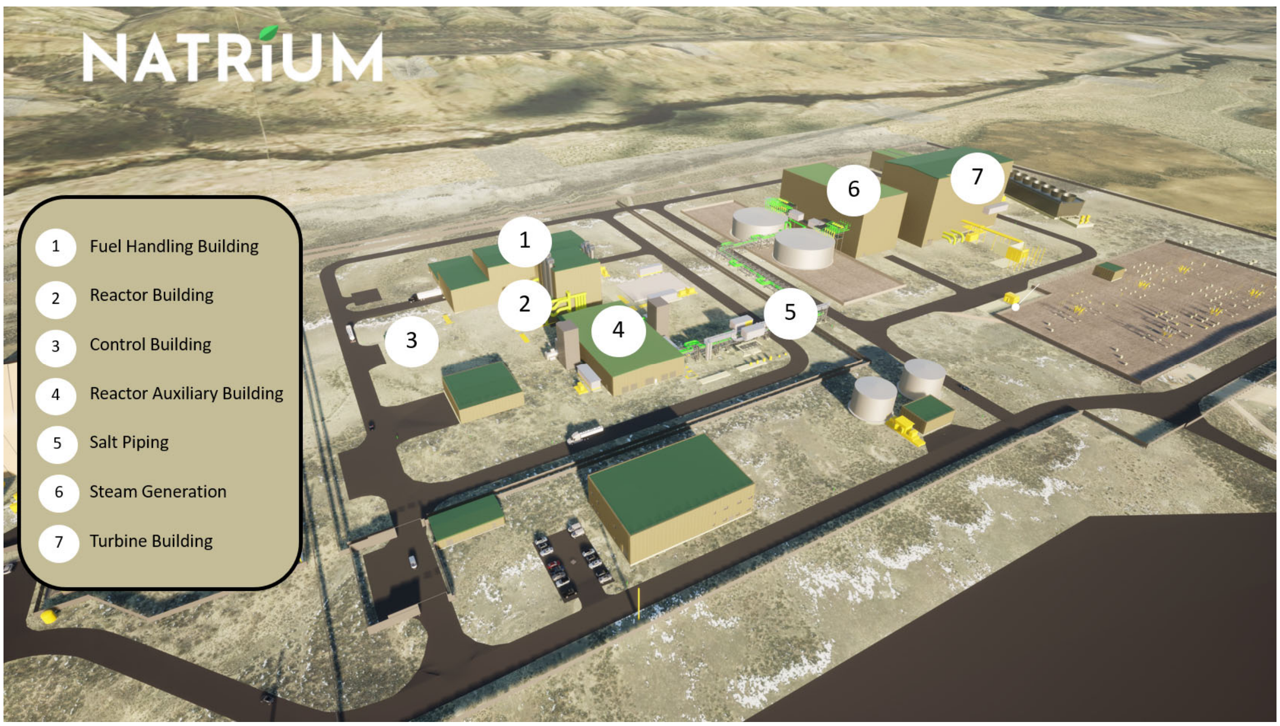

Figure 1-2. Plant Layout

The Natrium plant uses a pool-type design with the reactor core and primary coolant pumps located within a large pool of primary sodium coolant and no penetration through the RV thereby eliminating loss of coolant accidents (LOCAs) involving primary pumps and piping. The primary sodium pool operates at near atmospheric pressure. Heat is transferred from the hot primary sodium pool to an intermediate sodium piping loop by means of two Intermediate Heat Exchangers (IHXs). The intermediate piping loop uses sodium to transport reactor heat from each IHX to two Sodium-Salt Heat Exchangers (SHXs). These SHXs in the Nuclear Island heat salt received from the cold salt tank in the Energy Island. The heated salt is then returned to the Energy Island for storage in the hot salt tank, which serves as thermal energy storage. The salt stored in the hot tank is used to generate steam for use in steam turbine generators, eliminating the need for generating steam directly from reactive sodium metal. The Natrium plant can vary its supply of energy to the grid through its energy storage system. The Natrium reactor operates at a thermal power of 840 MW while the plant produces 336 MWe steady-state and 500 MWe peak power. Sample plant parameters are summarized in Table 1-1.

Table 1-1. Sample Plant Parameters

Parameter	Example Values
Reactor Type / Reactor Coolant	Fast neutron spectrum / liquid metal sodium
Heat Transport Architecture	Primary sodium pool → intermediate sodium loop → nitrate salt energy storage loop, superheated steam w/ reheat
Reactor Thermal Power	840 MWth
Electric Power Output	336 MWe steady state and up to 500 MWe peak
Energy Storage Capacity	[[ ]]$^{(a)(4)}$
Primary Operating Pressure	~Atmospheric

Metal fuel has been selected for the Natrium reactor based on high technology readiness demonstrated by EBR-II and FFTF. The reactor has been designed to accommodate both Type 1 and Type 1B fuel designs without modification of reactor internals. As such, fuel can be transitioned to Type 1B fuel when it is available. The initial loading and first few years of operation will utilize Type 1 sodium-bonded metallic U-Zr fuel.

The thermal energy storage system, located in the Energy Island, uses two molten salt tanks, one hot and one cold. Its architecture is like molten salt systems for concentrated solar power. The charging salt loop transports salt from the cold tank to the Nuclear Island for heating and routes it to the hot tank. The steam / salt loop transports salt from the hot tank to steam generators to generate superheated steam and returns salt to the cold tank.

The Natrium plant has been designed to accomplish reactivity control with multiple layers.

The non-safety-related reactor control system acts as a buffer to prevent the need for a scram. It detects abnormal operation and initiates a runback via motor driven insertion of neutron absorbing control rods to achieve a softer shutdown than a scram.

The SR reactor protection system initiates a scram if the reactor control system fails, or a runback fails to prevent the reactor from reaching a scram setpoint. The high reliability scram function is initiated by removing electrical power to an electromagnet, resulting in insertion of all control and standby rods into the reactor core.

The reactor core is designed with a negative temperature and power coefficient that is strong enough such that the reactor can accommodate anticipated transients without scram for events such as loss of primary flow, loss of heat sink, and uncontrolled rod withdrawal.

The high boiling point of sodium allows reactor operation at atmospheric pressure. A close-fitting Guard Vessel (GV) stops the loss of coolant should the RV develop a leak. Furthermore, the reactor cover gas operates at essentially atmospheric pressure so there is little driving force for a release.

The Natrium plant is designed to accomplish residual heat removal with multiple layers of protection.

Forced flow heat removal via Intermediate Air Cooling (IAC) serves as the normal shutdown cooling system for outages. There are two trains, one for each primary heat exchanger. The IAC has two cooling modes: forced flow and passive flow. For the final heat sink, it transfers heat to the atmosphere from the Sodium-Air Heat Exchangers (AHXs). Simple operation of a fail-open electromagnetic damper initiates passive cooling. Active operations support normal controlled cooling operations (such as during a refueling outage) and in response to anticipated transient events. Forced flow is provided by air blowers and the Intermediate Sodium Pumps (ISPs). The IAC's natural draft arrangement permits passive operation of the system as a diverse alternative if power to support forced cooling is not available. These functions supplement the SR Reactor Vessel Air Cooling (RAC) system and, as a result, enable the IAC and its support system designs to be non-safety-related.

The RAC removes decay heat using natural circulation of air around the exterior of the GV. The RAC does not have any dampers. RAC is always operating and requires no power, people, or control action to perform its function. The RAC relies on the natural circulation performance of the primary sodium and conductive/convective heat transfer to the RV wall. Thermal radiation heat transfer then dominates heat transfer to the GV. Natural draft air inlets provide ambient outside air to cool the GV wall via a combination of radiative and convective heat transfer.

The Nuclear Island is composed of six major buildings: reactor, fuel handling, control, electrical, reactor auxiliary, and fuel auxiliary buildings. The RXB, see Figure 1-3, houses two major components: the reactor and RAC air ducts. The reactor is located below grade to protect it from natural hazards (e.g., earthquakes, tornadoes, etc.) and other hazards. There are only two rooms in the RXB, the refueling access area, where refueling and maintenance takes place, and the head access area where limited maintenance takes place. Intermediate sodium piping exits the RXB below ground to the reactor auxiliary building.

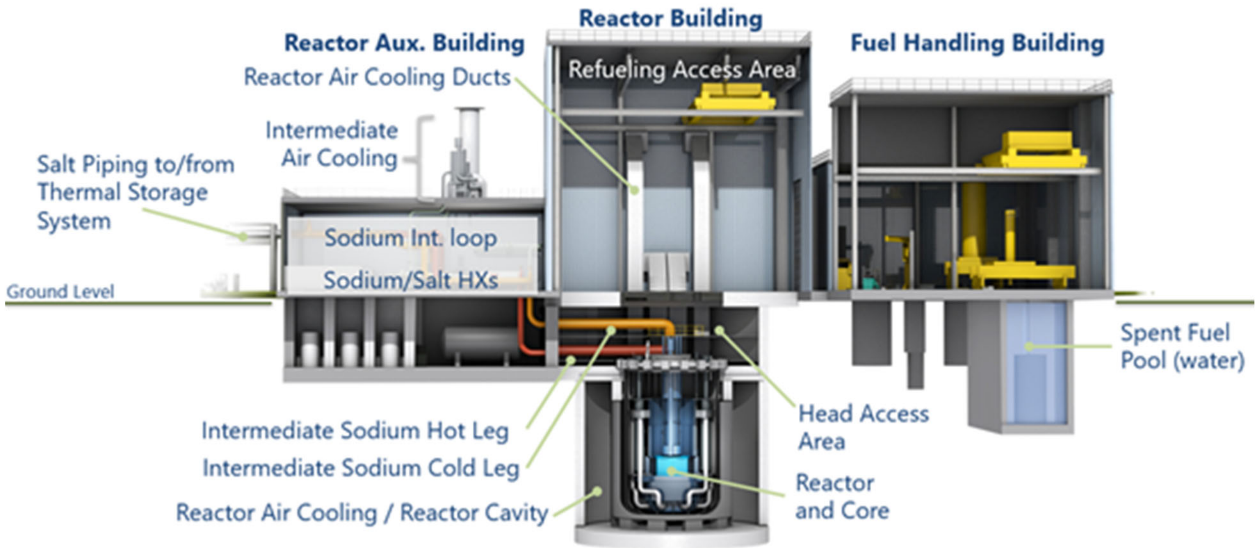

Figure 1-3. Elevation View

The Primary Heat Transport System (PHT) is contained within the RV and consists of the reactor core, the IHX, the PSPs, the hot pool, and the cold pool. The PHT sodium flows up through the core where the fuel assemblies heat the sodium. The hot sodium enters the hot pool and flows downward through the shell side of the two IHXs. The sodium, cooled by the IHT sodium coolant, exits the bottom of the IHXs and enters the cold pool. Cold pool sodium flows downward to the PSP inlet plenums which are located very near the bottom of the vessel to maximize coolant inertia. The PSPs drive the cold pool sodium downward from the inlet and discharge it into a series of core supply pipes, which return the sodium to the core inlet. The sodium then enters the core through the core support and distribution structure completing this flow circuit.

The Fuel Handling Building (FHB) houses fuel receipt equipment, refueling equipment, fuel storage equipment, and the fuel storage pool. Casks are used to transport fuel and in-reactor components from the RXB to the FHB. The buildings are connected by a rail system at ground level to support movement of the fuel handling cask. The FHB also contains the mechanical handling equipment which moves assemblies and provides access to the fuel pool. A bridge crane supports movement of dry storage fuel casks and equipment within the facility.

The Reactor Vessel Head (RVH) supports and locates the rotating plug for refueling operation. This plug is essential for the initial fueling of the reactor and for all subsequent fuel transfer operations during refueling and decommissioning. The plug is configured such that the In-Vessel Transfer Machine (IVTM) can access all core components, the In-Vessel Storage (IVS) locations, and the fuel elevator. The plug rotates via a bearing and drive assembly and is equipped with sealing mechanisms to isolate the primary fluid and cover gas from the atmosphere during normal, accident, and refueling operations. The GV surrounds the RV and is designed to contain sodium leakage in the event of an

RV breach, ensuring sufficient coolant inventory is maintained in the RV for residual heat removal through level equalization and preventing a sodium reaction with the surrounding RXB concrete.

The IVTM moves core assemblies between the core, in-vessel fuel storage racks, and transfer station for removal from the RV. It is mounted on the reactor rotatable plug, which is centered within the reactor top plate. The IVTM consists of two subassemblies: the above-head drive assembly and the in-vessel fuel handling mechanism. The latter extends to reach all removable core assembly locations when used in conjunction with the rotatable plug. Core assemblies are transferred into and out of the RV with the fuel transfer lift operating through the reactor transfer adapter. Fresh core assemblies are transferred through the cover gas space into the fuel transfer lift in the top of the pool region, and then lowered to core level to be transferred into the core using the IVTM. Used core assemblies are transferred out of the core to the IVS for decay or directly to the fuel transfer lift for assemblies which do not require in-vessel decay.

The ex-vessel fuel handling system components transfer all new reactor core assemblies from the point of receipt from the supplier through inspection and conditioning to the RV. On the back end of the reactor outage cycle, the ex-vessel fuel handling components take off loaded irradiated core assemblies to the EVST. Following the outage, offloaded assemblies in the EVST are processed through a Pool Immersion Cell (PIC) to a used fuel storage pool. The PIC provides the sodium residue removal allowing the assemblies to be stored in water for operations such as waste consolidation for non-fuel assemblies and underwater cask loading for used fuel assemblies. When desired decay heat limits are reached for used fuel assemblies they are processed into conventional dry casks and transferred to site storage pads for interim dry storage.

The water pool fuel handling system contains the equipment and structures needed to load, store, and retrieve irradiated core components and used fuel assemblies from the Used Fuel Pool (UFP). After the core components or fuel assemblies have had the sodium residue removed and have been immersed in water, the water pool fuel handling machine moves the core components or used fuel assemblies to the UFP. In the UFP, the core components or used fuel assemblies undergo long term decay before being removed using a cask.

The fuel transport and storage system packages and transports irradiated core components and used fuel assemblies for long term dry storage. It consists of the cask transporter and the interim dry storage pad. The dry cask transporter navigates to the cask transporter pickup location where the water pool fuel handling system has prepared and staged the dry storage cask for pickup.

1.4 Safety System Classification

The Natrium plant uses three safety classification levels: Safety-Related, Non-Safety-Related with Special Treatment (NSRST), and Non-Safety-Related with No Special Treatment (NST). Explanations for each of the three classifications are provided below.

1.4.1 Safety-Related

SSCs selected from the SSCs that are available to perform the Required Safety Functions (RSFs) to mitigate the consequences of Design Basis Events (DBEs) to within the LBE frequency-consequence (F-C) target, and to mitigate DBAs that only rely on the SR SSCs to meet the dose limits of 10 CFR 50.34 using conservative assumptions.

SSCs selected from the SSCs that are available and relied on to perform RSFs to prevent the frequency of BDBE with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C target.

1.4.2 Non-Safety-Related with Special Treatment

Non-safety-related SSCs relied on to perform risk-significant functions. Risk-significant SSCs are those that perform functions that prevent or mitigate any LBEs from exceeding the F-C target or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs. Non-safety-related SSCs relied on to perform functions requiring special treatment for DID adequacy. These SSCs are safety-significant even if they are not risk-significant.

1.4.3 Non-Safety-Related with No Special Treatment

All other SSCs (with no special treatment required).

1.5 Source Term Evaluation Model Accident Sequence Spectrum

Accident sequences evaluated within the PRA identify three categories of LBEs: Anticipated Operational Occurrences (AOOs), DBEs, and BDBEs. Events that are beyond the boundaries of the LBE release frequency range outlined within NEI 18-04 are identified as Other Quantified Events (OQEs). The events are categorized by frequency, consistent with the guidance outlined in NEI 18-04, as follows:

- AOOs are events with mean frequencies of $1x10^{-2}$ / plant year or greater
- DBEs are events with mean frequencies from $1x10^{-4}$ / plant year to $1x10^{-2}$ / plant year
- BDBEs are events with mean frequencies from $5x10^{-7}$ / plant year to $1x10^{-4}$ / plant year
- OQEs are events with a mean frequency below $5x10^{-7}$ / plant year

Additionally, DBAs are identified. Unlike the other LBEs, DBAs are not categorized by a mean frequency and are instead derived from DBEs by only crediting SR SSCs.

The Source Term EM is used to evaluate AOOs, DBEs, BDBEs, OQEs, and DBAs that involve release of radioactive material. If the AOO, DBE, BDBE, OQE, or DBA does not involve release of radioactive material, use of the Source Term EM is unnecessary.

2 EVALUATION MODEL CAPABILITY REQUIREMENTS

A four-step process was undertaken to define the capabilities of the Source Term EM. These steps included:

1. Specify analysis purpose, transient class, and power plant class (Section 2.1)
2. Specify FOMs (Section 2.2)
3. Identify systems, components, phases, geometries, fields, and processes that must be modeled (Section 2.3)
4. Identify list of important key phenomena (Section 2.4)

2.1 Analysis Purpose, Transient Class, and Power Plant Class

As the first step in an EM development, requirements and capabilities are established by specifying: the purpose of analysis of the source term, important phenomena and processes in transient and accident scenarios, and description of the Natrium plant.

The purpose of the Source Term EM is to provide the methodology for analyzing the MST for postulated releases from the Natrium plant. The transient classes considered are essentially all encompassing of those that are plausible, with emphasis on those scenarios that would yield potential fuel failure. Table 3-1 of NEI 18-04 [2] describing the LBE definitions is repeated here as shown in Table 2-1 for ease of reference and information purposes.

Table 2-1. Licensing Basis Events Definitions

Event Type	Guidance Document Definition
Anticipated Operational Occurrences	Anticipated event sequences expected to occur one or more times during the life of a nuclear power plant, which may include one or more reactor modules. Event sequences with mean frequencies of 1×10^{-2}/plant-year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the plant, regardless of safety classification.
Design Basis Events	Infrequent event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than AOOs. Event sequences with mean frequencies of 1×10^{-4}/plant-year to 1×10^{-2}/plant-year are classified as DBEs. DBEs take into account the expected response.
Beyond Design Basis Events	Rare event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than a DBE. Event sequences with mean frequencies of 5×10^{-7}/plant-year to 1×10^{-4}/plant-year are classified as BDBEs. BDBEs take into account the expected response of all SSCs within the plant regardless of safety classification.
Design Basis Accidents	Postulated event sequences that are used to set design criteria and performance objectives for the design of SR SSCs. DBAs are derived from DBEs based on the capabilities and reliabilities of SR SSCs needed to mitigate and prevent event sequences, respectively. DBAs are derived from the DBEs by prescriptively assuming that only SR SSCs are available to mitigate postulated event sequence consequences to within the 10 CFR 50.34 dose limits.
Licensing Basis Events	The entire collection of event sequences considered in the design and licensing basis of the plant, which may include one or more reactor modules. LBEs include AOOs, DBEs, BDBEs, and DBAs.

The types of events with potential for RN release either from fuel failure or loss of integrity of a system carrying primary coolant and the resultant release of dissolved isotopes have been identified. It is possible that the following phenomena could lead to fuel failure:

- Significant reduction in flow
- Localized high power-to-flow conditions
- Physical damage to the fuel (e.g., physical damage due to stochastic cladding failures during normal operation and potential impact damage due to scenarios like drop events)
- Reactivity insertion such as uncontrolled control rod withdrawal

Even though Step 1 of the EMDAP highlights transient classes, fuel failure and radioisotope buildup mechanisms are also considered for normal operation.

2.2 Figures of Merit

This section addresses EMDAP Step 2. FOMs need to be specified in this step by considering the following items: (1) FOMs are quantitative standards of acceptance that are used to define acceptable answers for a safety analysis, and (2) during EM development and assessment, a temporary "surrogate" FOM may be of value in evaluating the importance of phenomena and processes.

Two primary FOMs were selected for the source term phenomena identification and ranking table (PIRT) process (which is further described in Section 2.4). These FOMs for the PIRT are associated with a release scenario (e.g., fuel failure) or final radiological consequence (e.g., expected doses at the exclusion area boundary (EAB)). Dose potential is used to compare the relative importance of each phenomenon.

2.2.1 Inhalation Dose Potential

This FOM is defined to be used as a surrogate of the inhalation dose. The inhalation dose is a primary concern for an individual at on-site or off-site while breathing air that carries RNs.

2.2.2 Submersion Dose Potential

This FOM is defined to be used as a surrogate of the air submersion dose. The air submersion dose is typically from a cloud of noble gases to an individual as well as gamma and/or beta shielding concerns for an individual or equipment.

2.3 Systems, Components, Phases, Geometries, Fields, and Processes

This section addresses EMDAP Step 3. The purpose of this step is to identify the EM characteristics based on hierarchical system decomposition methods. An important principle to note is that if a deficiency exists at a high level, it is usually not possible to resolve it by fixing ingredients at lower levels. For relatively simple transients, the decomposition process will also be simple.

Pertinent, but not necessarily all, systems for source term evaluations are listed below. The design of fuel, vessel, and coolant systems are pertinent in determining the types of failure modes, the potential activation products outside the fuel, and the flow path of released material from potential fuel rod failure during normal and accident operations. The decontamination systems are also included in the subsequent systems listing because the systems filter/resins are sources of exposure during normal and accident conditions and they impact the intensity of the potential liquid (i.e., coolant) and gas (i.e., effluent) source terms.

2.3.1 Systems

2.3.1.1 Reactor Core Component System

The reactor core is designed as a fast reactor cooled by liquid sodium. The coolant flows upward through the core which is composed of fuel, control, reflector, shield, and standby assemblies. The fuel assembly produces heat and provides the neutron flux environment. Initial operation of the plant will consist of Type 1 fuel featuring a U-10Zr fuel column with a sodium bond to HT9 cladding as shown in Figure 2-1. Later the plant may transition to Type 1B fuel. The source term methods can adapt to these changes, e.g., fuel design, nuclear design, and system analysis, based on a fuel design change from Type 1 to Type 1B. Currently, there is not expected to be much difference in the overall Source Term EM with Type 1B fuel as the Source Term EM has been developed to be flexible for such a potential fuel transition. However, note that this report is only applicable to the Type 1 fuel.

(a)(4)

Figure 2-1. Type 1 and Type 1B Fuel

2.3.1.2 Reactor Enclosure System

The Reactor Enclosure System (RES) contains and supports the reactor core and primary sodium coolant, including all supporting equipment and structures. The RES is divided into five subsystems: RV, Reactor Internals (RI), RVH, GV, and Reactor Support Assemblies (RSA). All subsystems are in, and are either directly or indirectly supported by, the RXB. The RV, along with the RVH, form most of the reactor coolant and primary cover gas boundaries. Additionally, the RVH locates and supports extra-system equipment interfacing with the core and primary coolant. Finally, the RVH and RV provides support for the RIs as well as the Core Support Structure, which supports the reactor core.

2.3.1.3 Primary Heat Transport System

The PHT is entirely contained within the RV and consists of the reactor core, the IHXs, the PSPs, the hot pool, intermediate pool, and the cold pool.

2.3.1.4 Intermediate Heat Transport System

The IHT transfers heat from the PHT to the Nuclear Island Salt System (NSS). The IHT performs this function during normal power operation, startup, shutdown, and transient conditions. There are two IHT piping loops for each reactor module. Each intermediate loop is thermally coupled to the reactor PHT by an IHX. IHT non-radioactive sodium is circulated via the ISPs which transport heat from the IHXs to the SHXs. The IHT pumps are located in the cold leg to reduce their operating temperature. The main components of the IHT loops are the ISPs, SHXs, intermediate sodium hot and cold leg piping, expansion tanks, and the sodium drain tank.

2.3.1.5 Reactor Vessel Auxiliary Cooling System

The RAC is the SR residual heat removal cooling system for the reactor. It supplies natural draft outside ambient air for reactor cooling. The RAC system relies on the natural circulation performance of the primary sodium and the conductive/convective heat transfer to the RV wall. Thermal radiation heat transfer then dominates heat transfer to the GV. From there, natural draft air inlets provide ambient outside air to cool the GV wall via a combination of radiative and convective heat transfers. RAC is always in operation by nature of its open passive design, and therefore does not require equipment alignment, power, operator action, or support systems to perform at peak performance. As a result, RAC provides the SR decay heat removal system used to protect the fission product boundaries of the reactor through the most severe spectrum of plant events.

2.3.1.6 Intermediate Air Cooling System

IAC serves as the normal shutdown cooling system for outages. It has two cooling modes: forced flow and passive flow. There are two trains, one on each IHT loop. For the final heat sink, it transfers heat to the atmosphere from the AHXs.

2.3.1.7 Decontamination Systems

2.3.1.7.1 Sodium Processing System

The Sodium Processing System (SPS) is an auxiliary system. Its main purposes are to control and monitor reactor sodium chemistry. For the project, the SPS has two parts: Primary - SPS.1 and Secondary - SPS.2. SPS.1 provides purification functions for the primary sodium in the RV. SPS.2 provides purification functions for the intermediate sodium, which is a heat transfer loop between the primary sodium and the energy island's salt system.

2.3.1.7.2 Liquid Radioactive Waste System

The Liquid Radioactive Waste (RWL) system is designed to collect, segregate, process, sample, and monitor the non-sodium/non-salt liquid radioactive waste for recycle and/or discharge. The RWL tanks receive and store radioactive or potentially radioactive liquid waste. The RWL tanks store the waste during normal operation and during AOOs.

The RWL system is comprised of components such as tanks, pumps, and skid mounted equipment that could be potential sources.

2.3.1.7.3 Gaseous Radioactive Waste System

The RWG system collects, handles, and disposes of gaseous radioactive waste. It provides the capability for continuous treatment of the reactor cover gas.

The RWG system is designed to:

- Collect gaseous radioactive waste during all normal modes of operation
- Hold gaseous radioactive waste to allow for radioactive decay
- Filter and treat gaseous radioactive waste
- Dispose of gaseous radioactive waste via controlled release

It provides protection to plant personnel and the environment, minimizes radioactive releases and the spread of contamination, and ensures personnel exposures are As Low As Reasonably Achievable (ALARA).

The RWG system is comprised of various components such as tanks, charcoal adsorber beds, air filters, and mechanical equipment that could be potential sources.

2.3.1.7.4 Solid Radioactive Waste System

The Solid Radioactive Waste (RWS) system collects and packages solid radioactive material for offsite disposal. Spent fuel and major components which have been removed and replaced are not processed by the RWS system but are treated separately.

The RWS system is designed to:

- Collect solid radioactive waste from various systems produced during power and shutdown/outage conditions (spent fuel and major components which have been removed and replaced are treated separately and are not processed by the RWS system)

- Process solid radioactive wastes (for dewatering, stabilization, volume reduction, size reduction)
- Package solid radioactive wastes in containers, which are compliant with regulatory and disposal site requirements
- Store packaged solid wastes prior to removal from site
- The system containers serve as the barrier to the release of radioactive material during power, shutdown and outage conditions

2.3.2 Phases

The phases of sodium in LBEs of interest are liquid sodium and gas. The argon cover gas medium is in a gaseous phase. Phases considered for released RNs may include gas, aerosols, and particulates. RNs may be transported through liquid sodium, gaseous argon, air, and/or liquid water depending on the type of event being considered. [[

]](a)(4) Releases from failed fuel in the long term spent fuel pool would be treated as a typical LWR spent fuel pool release with releases from the pool water.

2.3.3 Geometrical Configurations (Phase Topology or Flow Regime)

The geometrical configuration of sodium in LBEs of interest is a single-phase liquid. The liquid sodium flows upwardly, downwardly, and horizontally. The argon gas is basically in a stagnant condition and is compressed or expanded depending on system pressure. Argon flows in the annular gap between the RV and the GV, air flows along the outside of the GV. The geometrical configurations of heat structures are Cartesian and/or cylindrical.

2.3.4 Transport Processes

There are many mechanisms that determine the transport of and interactions between constituent phases throughout the system. The following transport and interaction mechanisms are considered:

- Transport properties defining inter-nodal mass, momentum, and energy of liquid sodium
- Transport properties defining inter-nodal mass, momentum, and energy of argon gas
- Properties defining inter- and intra-nodal momentum transport between liquid sodium and argon gas
- Momentum interactions between liquid sodium and argon gas
- Momentum interactions with internal structures and surfaces of RV and GV
- Properties defining inter- and intra-nodal energy transport between liquid sodium and argon gas plus properties defining inter-nodal energy transport between liquid sodium and air through heat structures
- Properties defining intra-nodal energy transport between constituents and heat structures
- Properties defining energy production through fission and decay heat, including neutron kinetics

2.3.5 Functional Containment

Functional containment for the reactor during operation and accident conditions is provided by the GV. The GV envelopes the RV with an annulus around the RV cylinder and a gap under the RV bottom. The GV is bolted to the RVH and sealed by welding. The RV and the GV are supported by the RVH. The RVH is supported by the RXB structure via RSAs. The annulus between the RV and the GV is sized to retain the primary sodium in the unlikely event of a RV leak such that the reactor core, stored spent fuel, and inlets to the IHXs remain covered with sodium. There are no penetrations in the GV below the GV flange. Argon pressure is maintained at a constant level and is continuously monitored with pressure sensors. The RV-GV annulus sodium ionization detectors and sodium liquid detectors provide for early warning of any leak in the RV. The GV is specified to be leak tight and is filled with argon gas at a pressure above the reactor cover gas pressure.

During the fuel handling process the functional containment also consists of the boundaries provided by the following compartments.

- Ex-Vessel Handling Machine (EVHM)
- Ex-Vessel Storage Tank
- Bottom Loaded Transfer Cask
- Pin Removal Cell
- PIC
- Spent Fuel Pool
- Dry Cask Storage

The functional containment for the SPS also consists of cold traps, a cesium trap, and lined cells with inert atmospheres in which the traps are contained in addition to the Sodium Cover Gas Systems (SCG).

2.3.6 Fuel Handling Building

The FHB contains most of the fuel handling equipment and most refueling processes are conducted within this structure. While the reactor is operating new fuel and non-fuel components are received, inspected, prepared for RV introduction (inerted and preheated), and placed into the External Vessel Storage Tank in this building. During the refueling outage new core components are taken to the RXB and sodium wetted spent fuel and irradiated core components, both of which have very high gamma radiation levels, come from the RV into the FHB inside the EVHM inerted and shielded cask enclosure. The sodium wetted irradiated core components have a short-term residence in the EVST until they are moved to the PIC using the Bottom Loading Transfer Cask (BLTC), which is argon inerted and preheated. The sodium residuals are passivated in the PIC prior to entry into a traditional spent fuel water pool.

Following additional storage time, the irradiated core components are placed into spent fuel or waste casks, the casks are closed and packaged for storage and shipment prior to leaving the FHB. A large commercial spent fuel cask transporter will be used to move the Interim Spent Fuel Storage Casks from the FHB. There are additional large and small refueling components that are handled by cranes and transporters in the FHB. A refueling control room is located on the FHB main floor. Operators will be stationed in this control room when most refueling activities are in progress.

The RWG processing system is also located in the FHB. Apart from the RWG potentially radiative gas compressors, the system components will be in shielded and closed shielded cells with no routine personnel access. Personnel access into the RWG cells will be performed using confined space entry procedures. Some maintenance activities will be performed in the FHB, primarily on refueling equipment components, some of which will have internal radioactive sodium (with cesium of varying amounts following failed fuel operation) contamination. Each maintenance activity will be carefully planned to minimize contamination spread and some activities will be performed in portable contamination control enclosures. There will be inert gas (primarily argon) piping in many FHB locations to support refueling and RWG systems.

2.4 Identification and Ranking of Phenomena and Processes

This section addresses EMDAP Step 4. The principal product of the process is a PIRT.

A PIRT process has been conducted for representative radiological source term events of the Natrium design. Three postulated events - FHAs (AOO, DBE, DBA, or BDBE), SPS Leak (DBE, DBA, or BDBE), and Unprotected Loss of Flow with Degraded Pump Coastdown (ULOF+) (OQE) - were selected as representative radiological source term events. Phenomena and processes from these events are considered to cover the breadth of relevant phenomena and processes in other postulated events. One PIRT has been developed by internal and external subject matter experts for each of the selected three events (FHA, SPS Leak, and ULOF+).

The initial list of potential events was reviewed to identify events with potential for RN release either from fuel failure or loss of integrity of a system carrying primary coolant and the resultant release of dissolved isotopes. Based on industry experience, it is expected the phenomena listed in Section 2.1 have the potential to lead to fuel failure.

ULOF+ is selected as a representative event of which phenomena and processes are considered to cover most of the sequences of events leading to radiological consequence. ULOF+ is initiated by a loss of offsite power and loss of AC power. This automatically trips the primary pumps at full power, with pump coastdown system failing to operate normally. Here, the assumption is that multiple reactor protection systems fail to respond on demand, including a failure to scram, i.e., unprotected. A significant reduction of primary flow through the core leads to coolant boiling, clad failure, and fuel melting and relocations. Fuel melting is assumed to begin at the beginning of the transient depending on the severity of the transient initiating conditions, leading eventually to in-pin fuel relocation and or cladding failure, molten fuel ejection into the coolant channel, and ex-pin fuel and cladding relocation.

Furthermore, it was concluded that there are six events that are likely to require a dose consequence analysis:

- [[ ]](a)(4), Primary SPS Leak, Manual Shutdown Failure
- [[ ]](a)(4), Cover Gas Processing System Release Outside Containment
- [[ ]](a)(4), Fuel Drop During Insertion or Removal from RV
- [[ ]](a)(4), Fuel Drop Between EVST And Washing Station
- [[ ]](a)(4), Fuel Drop in Washing Station During Insertion Into PIC Or During Insertion Into Water Pool
- [[ ]](a)(4), Fuel Assembly Drop in Spent Fuel Pool

This list was subsequently simplified to FHAs and an SPS leak.

2.4.1 Phenomena Identification and Ranking Tables

Phenomena and processes that are relevant to radiological source term methodology are identified.

Importance rankings of the phenomena/processes identified are made according to a three-level scale shown in Table 2-2. This ranking assesses the level of modeling fidelity required to predict the FOMs reasonably well based on current knowledge of the phenomena. The importance ranking, therefore, may be regarded as the relative sensitivity of the FOM with respect to the expected variability about the expected values for the parameters associated with the phenomenon being considered.

Table 2-2. Phenomena/Processes Importance Rankings

Ranking	Description
High (H)	The sensitivity *(Note 1)* of FOMs to the phenomenon is large.
Medium (M)	The sensitivity of FOMs to the phenomenon is medium.
Low (L)	The sensitivity of FOMs to the phenomenon is little or negligible.

Note 1: The sensitivity of the FOM is with respect to the expected variability of the expected values.

Rankings of the knowledge level of phenomena/processes are made according to the three-level scale shown in Table 2-3. The knowledge level is determined in an absolute sense, independent of the associated importance ranking.

Table 2-3. Knowledge Level Rankings

Ranking	Description
High (H)	The phenomenon is well known. Data uncertainties are low and well characterized.
Medium (M)	The phenomenon is partially known. Data are available but the uncertainties are large.
Low (L)	There is little knowledge regarding the phenomenon. There are large uncertainties.

Rankings of importance and knowledge level of the phenomena and processes relevant to FHAs are presented in Table 2-4.

Rankings of importance and knowledge level of the phenomena and processes relevant to an SPS leak are presented in Table 2-5.

Rankings of importance and knowledge level of the phenomena and processes relevant to an ULOF+ are presented in Table 2-6.

Table 2-4. PIRT for FHAs

[[

Phenomena Number	Phenomenon / Process	Description	Importance Ranking	Rationale for Importance Ranking	Knowledge Level	Rationale for Knowledge Level

]](a)(4)

[[

Phenomena Number	Phenomenon / Process	Description	Importance Ranking	Rationale for Importance Ranking	Knowledge Level	Rationale for Knowledge Level

]](a)(4)

[[

Phenomena Number	Phenomenon / Process	Description	Importance Ranking	Rationale for Importance Ranking	Knowledge Level	Rationale for Knowledge Level

]](a)(4)

Table 2-5. PIRT for SPS Leak

[[

Phenomena Number	Phenomenon / Process	Description	Importance Ranking	Rationale for Importance Ranking	Knowledge Level	Rationale for Knowledge Level

]](a)(4)

[[

Phenomena Number	Phenomenon / Process	Description	Importance Ranking	Rationale for Importance Ranking	Knowledge Level	Rationale for Knowledge Level

]](a)(4)

[[

Phenomena Number	Phenomenon / Process	Description	Importance Ranking	Rationale for Importance Ranking	Knowledge Level	Rationale for Knowledge Level

]](a)(4)

[[

Phenomena Number	Phenomenon / Process	Description	Importance Ranking	Rationale for Importance Ranking	Knowledge Level	Rationale for Knowledge Level

]](a)(4)

Table 2-6. PIRT for ULOF+

[[

Phenomena Number	Phenomenon / Process	Description	Importance Ranking	Rationale for Importance Ranking	Knowledge Level	Rationale for Knowledge Level

]](a)(4)

[[

Phenomena Number	Phenomenon / Process	Description	Importance Ranking	Rationale for Importance Ranking	Knowledge Level	Rationale for Knowledge Level

]](a)(4)

Phenomena Number	Phenomenon / Process	Description	Importance Ranking	Rationale for Importance Ranking	Knowledge Level	Rationale for Knowledge Level
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						]](a)(4)

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Phenomena Number	Phenomenon / Process	Description	Importance Ranking	Rationale for Importance Ranking	Knowledge Level	Rationale for Knowledge Level

]](a)(4)

[[

Phenomena Number	Phenomenon / Process	Description	Importance Ranking	Rationale for Importance Ranking	Knowledge Level	Rationale for Knowledge Level

]](a)(4)

2.4.2 PIRT Results

Considering the combined results of the three PIRTs, Table 2-7 summarizes the phenomena which received importance-knowledge rankings of high-low, high-medium, and medium-low. Knowledge of these phenomena may not yet be appropriately developed considering their importance. This list will be used as a basis for scheduling further tasks, such as EM verification and validation, uncertainty quantification, etc.

Table 2-7. Summary of Higher Risk Phenomena

[[

Ranking	FHAs	SPS Leak	ULOF+

]](a)(4)

3 EVALUATION MODEL ASSESSMENT BASE DEVELOPMENT

This section addresses EMDAP Element 2.

3.1 Assessment Base Objectives

This section addresses EMDAP Step 5. The selection of the assessment database is a direct result of the requirements established in EMDAP Element 1. The database will include the following records: (1) separate effects tests, integral effect tests, benchmarks with other codes, plant transient data, and simple test problems, and (2) new experiments to validate the EM, if needed, based on the PIRT.

PIRTs have been developed by internal and external subject matter experts. Three scenarios were considered in the PIRT development. The selected scenarios are FHAs, SPS Leak/Rupture, and ULOF+. [[

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Table 3-1. Phenomena/Processes with High Importance Ranking

No.	Phenomena/Processes	Importance Ranking	Knowledge Level Ranking
1 [[			
2			
3			
4			
5			
6			
7			
8			
9			
10			
11			
12			
13			
14			
15			
16			
17			]](a)(4)

A database is established by acquiring appropriate experimental data relevant to the phenomena or processes listed in Table 3-1 to assess the requirements for the radiological Source Term EM. The first step in the assessment base development is to investigate the availability of legacy experimental data and evaluate the pedigree of the data. As part of future work, an assessment matrix will be developed, and testing needs will be identified to facilitate validation of the EM.

3.2 Scaling Analysis and Similarity Criteria

This section addresses EMDAP Step 6. Both top-down and bottom-up scaling analyses are conducted to ensure that the data, and the models based on those data, will be applicable to the full-scale analysis of plant transients. The optimum similarity criteria will be identified based on the important phenomena and processes identified in the PIRTs and the scaling analysis.

A database is established by acquiring appropriate experimental data relevant to the phenomena or processes listed in Table 3-1 to assess the requirements for the radiological Source Term EM. The first step in the assessment base development is to investigate the availability of legacy experimental data and evaluate the pedigree of the data. The next step is to develop an assessment matrix and identify testing needs for appropriate validation of the EM. For the last step, the available experimental data need to be gathered into the assessment matrix through digitization or procurement of the original dataset. In addition, in-house tests must be planned if the legacy tests do not cover all the high-ranked phenomena. The identification and pedigree evaluation of legacy tests and in-house test plans will be discussed in this section as part of a future revision to this report.

3.3 Existing Data Needed to Complete the EM Validation Database

This section addresses EMDAP Step 7. Based on the PIRTs, the assessment database will be filled with experiments and tests that best address the important phenomena and components. One example is an experiment being performed by the University of Wisconsin - Madison to evaluate the [[ ]](a)(4) It is not expected that a new experiment will be required to support the submittal of the preliminary safety analysis report of the Natrium plant. However, depending on the source term PIRT and the final design of the facility, additional experiments may be necessary to complete the EM assessment database. This section will be completed as part of a future revision to this report.

3.4 Evaluation of IET Distortions and SET Scaleup Capability

This section addresses EMDAP Step 8. The effects of the distortion from the IETs will be evaluated in this step. Correlations are based on SETs at various scales and the scaleup capability will be evaluated based on important phenomena and processes identified in the PIRT. This section will be completed as part of a future revision to this report.

3.5 Experimental Uncertainties Determination

This section addresses EMDAP Step 9. Uncertainties arise from measurement errors, experimental distortions, and other aspects of experimentation. Based on the experimental uncertainties, it will be determined whether the experimental data is qualified to be used in the model assessment. Discussions about how to evaluate uncertainties will be included when the uncertainties in the experiments (especially legacy experiments) were unknown or difficult to determine. This section will be completed as part of a future revision to this report.

4 EVALUATION MODEL DEVELOPMENT

EM development is guided by Steps 10 through 12 of RG 1.203 [1].

4.1 Evaluation Model Development Plan

This section addresses EMDAP Step 10. Based on the requirements established in EMDAP Element 1, an EM development plan is devised.

The EM Development plan identifies the necessary steps of the Source Term EM to be in accordance with the RG 1.203 Evaluation Methodology Development and Acceptance Process (EMDAP). The EM Development Plan task fulfills Step 10 in Element 3 of the EMDAP.

It is important to note that the Source Term EM development process considers the guidance on the EMDAP as described in RG 1.203. While adopting the LMP framework [2], endorsed as Regulatory Guide 1.233 [3], the Source Term EM development process does not meet verbatim conformance with RG 1.203 but rather considers the EMDAP as an industry best practice in methods development. Since the overall licensing strategy of the Natrium plant is pursuing a risk-informed, performance-based licensing framework, the source term methodology will be more realistic than previous LWR applications and will use an MST appropriate for a SFR.

An EM is a collection of calculational devices (codes and procedures) developed and organized to meet the requirements established in Element 1 of the EMDAP and described in the following sections of this report. Figure 4-1 provides a graphical representation of the Source Term EM and how it interfaces with other upstream and downstream methodologies.

(a)(4)

Figure 4-1. Source Term Evaluation Model Diagram

Appendix B of RG 1.203 provides an example showing the graded application of the EMDAP for an EM being revised. Although the Natrium Source Term EM is under development, this appendix of

RG 1.203 gives some useful insight into the steps required. An example is provided showing the stages of development for a small change to an existing calculational device. Section 4.2 of this report describes the structure of the calculational devices used for the Source Term EM. However, as the example in the RG lists the example in Step 10 of the EMDAP, this discussion is included in this section of the report.

The table from RG 1.203 Section B.1.3.1 is reproduced in the following subsections for each of the [[ ]](a)(4) calculational devices that are part of the Source Term EM. Note that some of the tasks, particularly the software V&V related tasks, are still in process.

The following computer codes are selected as calculational devices for the Source Term EM. Summary descriptions are also provided for computer codes used upstream to provide input and inform the Source Term EM.

4.1.1 [[ ]](a)(4)

[[

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Table 4-1. [[ ]](a)(4)

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]](a)(4)

4.1.2 [[ ]](a)(4)

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]](a)(4)

Table 4-2. [[ ]](a)(4)

[[

]](a)(4)

4.1.3 [[ ]](a)(4)

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]](a)(4)

Table 4-3. [[ ]](a)(4)

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4.1.4 RADTRAD

RADTRAD is selected due to its ability to calculate relevant dose consequences (RN decay), its availability, and high knowledge within the Natrium team. [[

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Table 4-4. RADTRAD Quality Assurance for Source Term EM

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4.1.5 Computer Codes Used Upstream of Source Term EM

The following computer codes are used to generate inputs that are employed in the Source Term EM.

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]](a)(4)

SAS4A/SASSYS-1 is a computer code developed by Argonne National Laboratory that is used for TH and safety analysis of power and flow transients in liquid metal cooled reactors. [[

]](a)(4)

4.1.6 Code Selection Gaps

An important aspect of the code selection process was the identification of any gaps in the ability of the selected codes to model important source term phenomena identified via the PIRT. The gaps that are not covered by the selected codes are listed as follows:

- [[

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It is important to satisfy analysis requirements related to these phenomena by using conservative assumptions, analysis defense-in-depth, and/or experimental results.

4.2 Evaluation Model Structure

This section addresses EMDAP Step 11. The EM structure includes the structure of the individual component calculation devices as well as the structure that combines the devices into the overall EM. It includes the structure of each individual calculation device (i.e., software code) as well as the structure combining the individual devices into the overall EM. Section C.1.3.2 of RG 1.203 [1] provides guidance on how to develop an EM structure.

The EM structure will be utilized in the development of analyses that will calculate the source term. The structure describes the following for each calculational device instrumental to the development of the source term: systems and components, constituents and phases, field equations, closure relations, numerics, and additional features. The EM structure also describes the output/input interfaces between each calculational device.

[[

]](a)(4) The description of interfaces between the codes are limited to those upstream and downstream of these [[ ]](a)(4) primary calculation devices. Output and input data exchange between the calculation devices will be passed through these interfaces in a controlled manner through automation or manually between analysis groups or team members via administrative controls.

Note that when SAS is used in this Source Term EM structure description it refers to the SAS4A/SASSYS-1 software.

4.2.1 [[ ]](a)(4)

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]](a)(4)

4.2.1.1 Systems and Components

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(a)(4)

Figure 4-2. [[ ]](a)(4)

4.2.1.2 Constituents and Phases

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]](a)(4)

4.2.1.3 Field Equations

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4.2.1.4 Closure Relations

[[

]](a)(4)

4.2.1.5 Numerics

[[

]](a)(4)

4.2.1.6 Additional Features

[[

]](a)(4)

4.2.2 [[ ]](a)(4)

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]](a)(4)

4.2.2.1 Systems and Components

[[

]](a)(4)

4.2.2.2 Constituents and Phases

[[

]](a)(4)

4.2.2.3 Field Equations

[[

]](a)(4)

4.2.2.4 Closure Relations

[[

]](a)(4)

4.2.2.5 Numerics

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]](a)(4)

[[

]](a)(4)

4.2.2.6 Additional Features

[[

]](a)(4)

4.2.3 [[ ]](a)(4)

[[

]](a)(4)

4.2.3.1 Systems and Components

[[

]](a)(4)

4.2.3.2 Constituents and Phases

[[

]](a)(4)

4.2.3.3 Field Equations

[[

]](a)(4)

4.2.3.4 Closure Relations

[[

]](a)(4)

4.2.3.5 Numerics

[[

]](a)(4)

4.2.3.6 Additional Features

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]](a)(4)

4.2.4 Calculational Device - RADTRAD

RADTRAD (RADionuclide Transport, Removal, And Dose code) is a software program originally developed by SNL for the NRC. Several versions of the code were branched off following versions 3.02 and 3.03. The Natrium project has selected to use RADTRAD 3.10. This version was developed by Alion Science & Technology, which is now Serco-NA. There is a User Manual [19] that describes how to interface with RADTRAD 3.10; however, much of the underlying models and code description is found in NUREG/CR-6604 [20] including Supplements 1 [21] and 2 [22]. Further use of RADTRAD in this document refers to RADTRAD 3.10 unless otherwise noted.

4.2.4.1 Systems and Components

RADTRAD represents the various physical regions being modeled primarily with compartments. The compartments can be generalized/normal or specific. Examples of the specific compartments include the control room and environment. The regions generally are considered to consist of the vapor phase since the code was developed for a release in LWR confinement buildings, although the code is basically phase-agnostic. That is, a source term is used that introduces the RNs to the space outside of the reactor; RADTRAD does not model RN release from the fuel - it must be entered by the user. The source term nuclide inventory file, release fraction and timing file, and dose conversion factors files are provided by the user as input. Compartments are connected with pathways which may have filter components placed on them.

4.2.4.2 Constituents and Phases

The working fluid that transports RNs is not modeled per se. RNs are the constituents of interest that are modeled in RADTRAD. They are introduced by a predefined or user defined source term. RADTRAD 3.10 has a limitation of only including 100 unique isotopes. Sprays can be used for nuclide removal, but the droplets are not modeled explicitly.

4.2.4.3 Field Equations

Conservation of RNs are the only significant field equations used by the code. There is no conservation of mass, momentum, or energy since the code does not explicitly model the fluid carrying the nuclides. The transport rate between compartments is user defined in units of cubic feet per minute. Section 2.1.1 of NUREG/CR-6604 provides the equation for conservation of RNs in the compartments.

4.2.4.4 Closure Relations

The equations for nuclide removal are provided in Section 2.2 of NUREG/CR-6604. These include the removal mechanisms of sprays, natural deposition, overlying pools, leakage, and filters. Most of these removal mechanisms are user defined. However, there are some built-in models such as the Powers model for sprays, Henry's correlation or Powers model for natural deposition, Powers model for bubble rise/pool scrubbing in an overlying pool, and the Brockmann and Bixler models for deposition in piping.

4.2.4.5 Numerics

Section 2.4 of NUREG/CR-6604 describes the solution method while the numerical solution technique specifically is given in Section 2.4.1. A Laplace transform method is used to solve the transport portion of the problem. The scenario time steps are defined by the input (i.e., a time step occurs every time there is a change in an input, which could be rather infrequent during a problem). There is some additional user control available over the time step size via supplemental time steps, and a time step size sensitivity study will be performed as part of an analysis using RADTRAD.

4.2.4.6 Additional Features

RADTRAD calculates the dose in a compartment, or the environment based on the user-input breathing rates, atmospheric dispersion factors (χ/Qs), and dose conversion factors.

4.2.5 Source Term EM Methodology Interfaces

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(a)(4)

Figure 4-3. Source Term Methodology Interfaces with Other Methodologies

4.2.6 Normal Operation

4.2.6.1 Normal Ongoing Effluents

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The selected software or manual calculations may be used to determine the RN inventory concentrations at various release points to the environment. These normal effluent source terms may be used by downstream evaluations to determine normal dose consequences to personnel and the public.

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]](a)(4)

Figure 4-4. Normal Operation Effluents EM Diagram

4.2.6.2 Spent Fuel Source Terms

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]](a)(4)

4.2.6.3 Primary Coolant due to Normal Fuel Defects

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]](a)(4)

With the nuclide source from the fuel determined, the selected code can be used to analyze the RN source term in the primary sodium coolant, presumably considering cleanup systems as well.

4.2.6.4 Activation Products Generated from Core Flux

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4.2.6.5 Buildup in Decontamination Systems & Waste Streams

The source term due to buildup of nuclides in the decontamination systems and waste streams may be analyzed in conjunction with the source term of the primary system coolant. [[

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(a)(4)

Figure 4-5. Decontamination Systems & Waste Streams EM Diagram

4.2.6.6 Personnel Operational Scenarios/ALARA

The source terms for ALARA personnel dose computations may be developed based on a combination of the other normal operation source terms.

4.2.7 System Leakage Scenarios Modeling Strategy

The system leakage scenarios are assumed during normal operation and not as part of, or consequence of, a different event. [[

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4.2.7.1 Radioactive Waste Gas System Leak

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Figure 4-6. RWG System Leakage EM Diagram

4.2.7.2 Sodium Cover Gas System Leak

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]](a)(4)

4.2.7.3 Sodium Processing System Leak

Like the SCG, the SPS source term will be based on the coolant inventory during normal operation. [[

]](a)(4)

(a)(4)

Figure 4-7. Sodium Cleanup System Leak EM Diagram

4.2.7.4 Intermediate Heat Transport System Leak

The source term development for the IHT leak will be evaluated in the same fashion as the sodium cleanup system leak. [[

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4.2.8 Licensing Basis Events (except for DBAs) Modeling Strategy

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(a)(4)

Figure 4-8. LBE Source Term EM Diagram

4.2.9 Design Basis Accident Modeling Strategy

Regarding the Source Term EM, the DBA EM structure is expected to be the same or very similar to the non-DBA LBE EM structure. Different analysis inputs and assumptions will be made, and certain features of the software may be utilized to model SR equipment to mitigate the accident, but the interfaces between the calculation devices/codes is expected to be the same. The DBA dose consequence is expected to be calculated with RRCAT.

4.2.10 Fuel Handling Accident Scenarios

There are a range of potential FHAs that could be considered. They can be classified into four primary categories based on where the accident occurs: in-vessel, in the washing station, in the spent fuel pool, or during transfer. This distinction is important since the fluid in which the accident occurs will affect the software to be used in the analysis.

It is presumed that all the FHA analyses will assume a particular fraction of the dropped fuel will be damaged with the RNs released over some time period. For traditional LWR analyses this is often assumed to be failure of 100% of the fuel rods in the dropped assembly with an instantaneous release of 100% of the gap activity.

4.2.10.1 FHA in-vessel

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Evaluation of an FHA in the EVST will be similar to, and likely bounded by, an in-vessel FHA. In-vessel FHAs involve more recently irradiated fuel assemblies than those considered with an ex-vessel FHA. As such, the ex-vessel FHAs are not expected to cause as significant of a safety hazard.

4.2.10.2 FHA in Spent Fuel Pool

The analysis of a fuel drop in the spent fuel pool is expected to be similar to that for an LWR. [[

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[[ ]]$^{(a)(4)}$

Figure 4-9. FHA in SFP EM Diagram

4.2.10.3 FHA in Washing Station

A fuel drop in the washing station may result in a breach of the fuel pin cladding. The EM structure is like that of the FHA in the spent fuel pool. [[

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(a)(4)

Figure 4-10. FHA in Washing Station EM Diagram

4.2.10.4 FHA during Fuel Transfer

This scenario covers a fuel drop between the vessel and the EVST and a fuel drop between the EVST and the washing station. The drop between the vessel and EVST is classified as an AOO and the drop between the EVST and washing station is classified as a DBE. Neither of these events are expected to need a source term generated as there are no pin cladding breach or failures expected per the event summaries.

4.2.11 Plume Exposure Pathway Emergency Planning Zone Sizing Methodology

The Source Term EM interfaces with the plume exposure pathway EPZ sizing EM . The source terms that are developed for various events and scenarios may be utilized as input in calculating doses to the public at the EPZ for both normal operation and accident scenarios.

4.2.12 Neutronics Methodologies

Explicit description of the neutronics methodologies are not part of the scope of the source term evaluation methodology. [[

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4.2.13 Dose Mapping for Equipment Qualification Evaluations

The normal operation source term may be used for EQ dose mapping. The steady-state RN inventory developed for the normal operation source term will be used as input to determine the relevant activity concentrations in portions of the primary coolant and supporting systems piping and equipment to calculate doses at various distances from those sources.

Post-accident source terms also will need to be determined for airborne RNs, post-accident cleanup systems (e.g., filters), piping, etc. The source terms developed for LBEs and DBAs will be able to be utilized for post-accident EQ beta and gamma dose evaluations.

4.2.14 Tritium

The Source Term EM will consider the sources and transport of tritium. [[
]](a)(4)

4.3 Closure Models

This section addresses EMDAP Step 12. Closure relationships or closure models describe a specific process during a plant transient and can be developed and/or incorporated in the principal analytical computer code, if needed. Closure models are mostly developed based on the results of SETs but may also rely on the results of IETs on rare occasions. The developed models need to be incorporated into the main analytical computer codes. If closure models are not developed, this step is skipped.

Closure relations for the specific calculational devices are listed in Sections 4.2.1 through 4.2.4. Strategies for developing models for the source term are described in this section. In particular, the functional containment modeling strategy and the RN transport modeling strategies are discussed.

4.3.1 Functional Containment Modeling Strategy

The Functional Containment Modeling Strategy provides a structured process that identifies the crucial but potentially evolving topics of the Natrium design, technology knowledge base, and licensing process, and recognizes and addresses their risks and potential impact on the overall functional containment design. The strategy defines the modeling investigations necessary to facilitate and direct that design evolution. The Containment Modeling Strategy defines the process that will be employed for each of the various RN release events or conditions. A strategy is recommended for each of the RN Release Event Categories to direct the development and evolution of the functional containment Modeling to address the specific issues, risks, and information gaps as they apply to each release Event Category.

The strategy directs the development of models to demonstrate the adequacy of the functional containment to perform its primary safety function to mitigate the on-site and off-site dose consequences to the acceptance limits established for the various events. As an integral part of that scope the modeling will include analyses to:

- Evaluate compartment environmental conditions (pressure, temperature, humidity) to be assessed against design limits of the compartment structures and barriers (i.e., barriers can reasonably be relied on for a confinement function)

- Determine or confirm pressure temperature dependent compartment leakage values to be used as input into RN transport calculations
- Evaluate compartment conditions from a sodium-chemical reaction.

The functional containment modeling strategy will provide a structured approach to drive the development of the functional containment models to support the objectives of Safety, Cost Effectiveness and Risk Management effectively and efficiently as the models evolve in response to the maturing plant design and increased understanding of the important phenomena. The primary elements of that strategy are listed as follows.

4.3.1.1 Event Categorization

Events that will require containment performance analysis can generally be thought of to fall into four high-level categories.

- Releases from In-Vessel Events
- Releases from Ex-Vessel Events
- Releases from Sodium Chemical Reactions
  - Primary Sodium Source
  - Secondary Sodium Source
- Normal Operation Releases and Effluents

4.3.1.2 Information Management / Risk Assessment

An essential element of a successful functional containment modeling strategy is the need to manage the evolving understanding of critical phenomena and maturing plant design inputs that are crucial to the development of a functional containment design and model. For each category of events a database will be maintained to track all assumptions used in the functional containment modeling and any action items assigned to resolve any assumptions or open issues.

4.3.1.3 Modeling Development / Evolution

Application of the functional containment concept to SFR events has little precedence regarding the identification and modeling of the phenomena that are important to performance of the containment during various radiological events to be analyzed.

Development of the SFR functional containment models is further complicated by the fact that SFR accidents and events are substantially different from LWR events where the containment response is driven by a major release of high temperature and pressure water and steam into the containment compartment resulting in an almost instantaneous and substantial pressure excursion within one or more of the primary containment compartments. For those analyses, the behavior of the containment in response to those events can generally be characterized by assumed perfect mixing of the release within the compartments and large pressure differentials within the containment compartments that are relatively easily defined by relatively straight forward pressure / flow calculations.

SFR events involve relatively minor leak rates from the vessel that do not provide sufficient mass and energy to ensure perfect mixing of the RNs in the compartment or create abrupt and significant pressures excursions that clearly dominate the containment response and drive the flow through the containment. Therefore, the leakage paths and flow rates from the various functional containment compartments will be driven by the active ventilation systems within those compartments. However, during events that include a loss of these active HVAC ventilation systems, the leakage paths and flow rates are driven by relatively subtle phenomena associated with the compartment heat load driven heat-ups and natural circulation paths within and between the compartments. The behavior of RN release within the functional containment compartments and the potential leakage rates from those compartments are not currently defined or readily represented by simple assumptions. Therefore, the SFR modeling strategy must start with basic modeling to investigate the response of the containment compartments under accident conditions and to identify the phenomena and parameters important to the mixing and leakage behavior during the event. The insights gained from this initial investigation will be used to refine the modeling to enhance its capabilities, and to model the important phenomena and demonstrate the sensitivities to various design input. It is expected that this "investigation/model enhancement" feedback loop will require a few evolutions to develop an acceptable model.

4.3.1.3.1 [[ ]](a)(4) Thermal/Hydraulic Modeling

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4.3.1.3.2 RN Transport / Dose Consequences

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]](a)(4)

4.3.1.3.3 Sodium Fire Modeling

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]](a)(4)

4.3.1.3.4 Initial Sensitivity Studies

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4.3.1.3.5 Model Evolution

As previously discussed, the strategy will maintain and update a data base of the status of crucial information that is expected to evolve during the functional containment design process. Based upon any changes in this information as well as the increased understanding of the functional containment behavior under accident conditions [[

]](a)(4), functional containment modeling is expected to evolve. The models will evolve to reflect the maturing design of the plant and refined information used in modeling the functional containment performance. [[

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4.3.1.4 Containment Modeling Task Structure

Due to the broad scope and evolving nature of the effort to develop the modeling of the functional containment, the project has been organized into several tasks and subtasks to facilitate its management. [[

]]$^{(a)(4)}$

4.3.2 Radionuclide Transport Modeling Strategy

The RN Transport Modeling Strategy defines the process that will be employed for each of the various RN release events or conditions.

4.3.2.1 Event Categorization

The event categorization for the RN transport modeling strategy is the same as that for the functional containment modeling strategy described in Section 4.3.1.1.

4.3.2.2 Radionuclide Mitigation Phenomena

The RN transport modeling strategy is predicated upon an understanding of the RN mitigation phenomena provided by each of the compartments making up the functional containment and the recognition of the effectiveness of those barriers on the various physical and chemical forms of the RN leakage being transported through the functional containment pathways to the eventual release to the environment. [[
listed below.

]]$^{(a)(4)}$

- [[

]](a)(4)

4.3.2.3 Radionuclide Groups

Generally, the potential constituents of the RN release can be grouped together based upon their similarities in chemical and physical characteristics that are important to assessing the behavior of that element as it is released from the fuel pellet itself or transported through the functional containment barriers and is released to the environment. [[

]](a)(4)

4.3.2.4 Radionuclide Transport Strategy

Effective modeling of the RN transport through the functional containment requires a quantitative assessment of the RN inventory in each of the successive functional containment compartments as the RN leaks or is driven to its eventual release to the environment. The modeling must also account for the effectiveness of the identified RN mitigation phenomena as the release is transported through the various functional containment compartments enroute to its eventual release to the environment. The modeling must also account for the dispersion of the released RN within the environment and its subsequent radiological dose impact of that RN as it reaches the public. [[

]](a)(4)

4.3.2.5 Modeling Development / Evolution

Development of an effective RN modeling strategy will begin with simple models employing low risk mitigation phenomena to establish some base line understanding of the radiological severity of the event and extent to which various mitigating phenomena will need to be credited to demonstrate successful dose consequences. As part of this strategy, the dose contributions from the individual RN forms (noble gases, particulate, halogens, and volatile RN) will be determined by scoping studies to better understand what mitigating strategies will be most effective in achieving acceptable doses. These studies will inform the evolution of the RN transport modeling to develop those aspects of the RN transport that provide maximum dose benefit while incurring limited risk that the modeling will be rendered ineffective due to closure of the gaps that currently exist in the understanding of the RN transport phenomena.

4.3.2.6 Initial Scoping Studies

The initial RN scoping studies will establish a general quantitative perspective of the effectiveness of the functional containment dose mitigation performance in response to the currently defined events and current preliminary state of knowledge about the critical parameters of the functional containment.

4.3.2.7 RN Transport Modeling Task Structure

The RN transport modeling task structure is similar that described for the functional containment modeling strategy described in Section 4.3.1.

4.4 General Conservative Methods

This subsection expands on the EMDAP Step 18 discussion. Conservative aspects have been identified for the following areas:

- [[

]]$^{(a)(4)}$

This subsection will also provide strategy for selection of conservative input parameters.

4.5 Event-Specific Methods

This portion of the methodology will be updated as analysis needs warrant as the Natrium design matures and any additional unique events are identified. The developed Source Term EM is applicable in full or in part to the following event types. The EM structure described in Section 4.2 of this report outlines the method for the event types that are listed below with example scenarios.

- LBEs
  - AOOs
    - [[

      ]]$^{(a)(4)}$
  - DBEs
    - [[

      ]]$^{(a)(4)}$
  - DBAs
    - [[

      ]]$^{(a)(4)}$
  - BDBEs
    - [[

      ]]$^{(a)(4)}$

- Normal operation
    - Normal ongoing effluents
    - Spent fuel source terms
    - Primary coolant due to normal fuel defects
    - Activation products generated from core flux
    - Buildup in decontamination systems & waste streams
    - Personnel operations scenarios/ALARA
- Plume exposure pathway EPZ sizing methodology
- Neutronics methodologies (including RN inventory generation)
- Dose mapping for EQ evaluations

Table 4-5 lists sources to be evaluated for normal operation and system leak scenarios [[ ]](a)(4)

Table 4-5. Normal Operation and System Leak Sources

Source Evaluated	Normal Operation Scenario/Purpose	[[
Fission Product Inventory	Spent Fuel Source Term Dose Mapping EQ	
Coolant Source Term	Dose Mapping EQ	
Decontamination System Buildup	Dose Mapping EQ	
Effluent	Dose Mapping EQ Personnel Exposure Potential EPZ Impact	
Normal Leak Scenarios Examples: RWG System Leak Cover Gas Cleanup System Leak Sodium Cleanup System Leak IHT System Leak	Dose Mapping Personnel Exposure	]](a)(4)

5 EVALUATION MODEL ADEQUACY ASSESSMENT

An EM has been developed and structured for consideration of potential source terms for the Natrium design. The source term methodology also covers RN releases during normal operation. Anticipated fuel defect and neutron activation could result in some potential minor periodic release of RNs to the environment. Hence, the source term methodology also includes radiological source terms for effluents, radwaste system design, shielding design, and EQ. The potential source terms developed span normal operations, system leakage scenarios, plausible accident scenarios, and emergency zone planning.

An additional aspect of assessing the adequacy of the proposed Source Term EM falls within performing a comparison of the Natrium methodology to RG 1.183 RP 2 (Attributes of an Acceptable Alternative Source Term) [4]. Although RG 1.183 is a guidance document, the regulatory positions noted in it have been determined to be relevant for evaluating the adequacy of a source term model. For example, the following discussion is provided in comparison to each Regulatory Position of RG 1.183. RP Section 2. Position bullets are addressed as follows and are relevant to the proposed adequacy of the Natrium Source Term EM:

1. RP 2.1 specifies that an alternate source term, (herein abbreviated as source term only) must be based upon major accidents for purposes of design analyses or consideration of possible potential accidental events. The guidance document also indicated that the source term must address events that involve a substantial meltdown of the core with subsequent release of appreciable quantities of fission products. The following sections within section 5 of this document describe the license basis events that have been evaluated for design analyses purposes. Also, potential design basis accidents, have also been evaluated relevant to identifying the source term, and the potential releases and release paths. The following paragraphs will describe those events, scenario types and describe how the model proposed is adequate for assessment of the source term.
2. RP 2.2 identifies that the source term must be expressed in terms of times and rates of appearance of radioactive species released and the chemical forms of iodine released. The following sections address utilization of a process for identifying the potential release in terms of times and rates of appearance through use of various code applications. Therefore, further discussion is provided related to the fact that appearance rates and releases will be addressed.
3. RP 2.3 states that the source term must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events. The RP further indicates that risk insights may be used, not to select a single risk-significant accident, but rather to establish a range of events to be considered. Relevant risk insights have been utilized regarding to establishing a source term model, such as using a PIRT based upon subject matter expert reviews. The identification of and review of a spectrum of credible severe accident events, and all other license bases events has been performed and documented. The PIRT process is discussed in Section 5.2.3 herein. The PIRT process identified high risk insights regarding the Source Term model and strategies for addressing such is ongoing. Related to reviewing a range of events this has also been documented and will be described in the sections to follow.
4. RP 2.4 identifies that a source term must have a defensible technical basis supported by sufficient experimental and empirical data, be verified, and validated. The adequacy of the proposed Source Term Model that is discussed in sections to follow, identifies that both experimental and empirical data have been utilized related to the computer codes that are planned to be utilized. Likewise, these proposed computer codes to establish the Source Term are undergoing internal verification and validation.

5. RP 2.5 identifies that a source term must be peer-reviewed by appropriately qualified subject matter experts. The peer-review and comments will be part of documentation supporting the source term. The assessment of the source term relevant to review and documentation has been conducted with subject matter experts, both internally and externally to the Natrium project team. Comments related to the PIRT have also been documented, and further reviewed for relevancy applications, i.e., high risk phenomena and uncertainty establishment.

Therefore, with respect to utilization of the RP guidance from RG 1.183 for considering attributes of a source term, this section will further provide justification for meeting the intent.

An initial source list was developed relevant to scenario types and ultimate potential releases. This initial list was used as the bases for further evaluating the Source Term EM.

Table 5-1 identifies the source to be evaluated, purpose and the ultimate end point of a potential release. This table ties together the potential sources that could be anticipated relevant to the Natrium design and is provided for information regarding establishing adequacy of the Source Term EM.

Table 5-1. Potential Initial Source List and Release

Potential Source Evaluated	Potential Scenario Type / Purpose	Potential Release / End Point
Fission Product Inventory	Normal and Accident Evaluations Spent Fuel Source Term EQ Dose Mapping	Spent Fuel Pool Spent Fuel Cask / Transport Cask Spent Fuel Decontamination System Buildup
Coolant Source Term	Normal Evaluations EQ Dose Mapping	System Coolant Piping Tanks Heat Exchangers Localized Leak Scenarios
Decontamination Systems Buildup	Normal Evaluation Dose Mapping EQ	Decontamination Resins Tanks
Fuel Handling Accident	Accident Scenario Personnel Exposure Potential EPZ Impact Accident Source for Specific EQ due to Decontamination System Buildup	Spent Fuel Pool Release from Building Spent Fuel Decontamination System Buildup
Effluent	Normal Scenario Dose Mapping EQ Personnel Exposure Potential EPZ Impact	Buildup within RXB Release from RXB Stack
Normal Leak Scenarios Expected Examples RWG System Leak Cover Gas Cleanup System Leak Sodium Cleanup System Leak IHT Leak	Normal Scenario Dose Mapping Personnel Exposure	Buildup within RXB Release from RXB Stack Localized Exposure at Point of Release
Primary SPS Leak	Accident Scenario System leak accident scenario at the location of the leak. Exposure to personnel EQ – Accident Potential Personnel Exposure	Localized Exposure at Point of Release
Cover Gas Processing System Release Outside Containment	Accident Scenario System leak accident scenario at the location of the leak onsite to personnel. System Leak Accident Scenario	Buildup within RXB Release from RXB Stack Localized Exposure at Point of Release

Potential Source Evaluated	Potential Scenario Type / Purpose	Potential Release / End Point
	EQ - Accident EPZ	
Fuel Drop During Insertion or Removal from RV	Accident Scenario EQ - Accident EPZ Personnel Exposure	Buildup in Decontamination System Buildup within RXB Release from RXB Stack
Fuel Drop Between EVST And Washing Station	Accident Scenario Contamination EQ - Accident Personnel Exposure EPZ	Local Surface Contamination Evaluations Buildup within RXB Release from RXB Stack
Fuel Drop-in Washing Station During Insertion Into PIC Or During Insertion Into Water Pool	Accident Scenario Contamination EQ - Accident Personnel Exposure EPZ	Local Surface Contamination Evaluations Buildup within RXB Release from RXB Stack
Fuel Assembly Drop in Spent Fuel Pool	Accident Scenario Contamination EQ - Accident Personnel Exposure EPZ	Local Surface Contamination Evaluations Buildup within RXB Release from RXB Stack
Loss of All Primary Pumps, No Failure	Accident Scenario EQ - Accident Qualification Release Scenario - EPZ	Local Surface Contamination Evaluations Buildup within RXB Release from RXB Stack Release into Coolant
Loss of All Primary Pumps, Runback Failure	Accident Scenario EQ - Accident Qualification Release Scenario - EPZ Personnel Exposure	Local Surface Contamination Evaluations Buildup within RXB Release from RXB Stack Release into Coolant
Cover Gas Processing System Release Inside Containment	Accident Scenario EQ - Accident Qualification Release Scenario - EPZ Personnel Exposure	Local Surface Contamination Evaluations Buildup within RXB Release from RXB Stack
Fuel Drop During In-Vessel Movement	Accident Scenario EQ - Accident Qualification Release Scenario - EPZ Personnel Exposure	Local Surface Contamination Evaluations Buildup within RXB Release from RXB Stack Release into Coolant

Potential Source Evaluated	Potential Scenario Type / Purpose	Potential Release / End Point
Fuel Drop Between RV and EVST	Accident Scenario EQ - Accident Qualification Release Scenario - EPZ Personnel Exposure	Local Surface Contamination Evaluations Buildup within RXB Release from RXB Stack Release into Coolant
Spent Fuel Assembly Is Crushed by EVHM Or BLTC Movement During Earthquake	Accident Scenario EQ - Accident Qualification Release Scenario - EPZ Personnel Exposure	Local Surface Contamination Evaluations Buildup within RXB Release from RXB Stack Release into Coolant
Spent Fuel Assembly Overheat	Accident Scenario EQ - Accident Qualification Release Scenario - EPZ Personnel Exposure	Local Surface Contamination Evaluations Buildup within RXB Release from RXB Stack Release into Coolant

A structure for calculation devices (i.e., software codes) and combining the use of individual devices into a Source Term EM is discussed within this TR. This section assesses the overarching Source Term EM with respect to adequacy but does not provide explicit details of calculated accuracies at this point in time.

Table 5-1 was further evaluated relevant to mechanisms attributed to their generation, timing and potential release.

There are many mechanisms that determine the transport and interactions of sources throughout a system. Transport and interaction mechanisms were categorized to be considered for further evaluation. [[ ]](a)(4) Those source term interactions that involve potential RN transport and release were identified to be evaluated with the codes [[ ]](a)(4) and RADTRAD (v. 3.10). [[ ]](a)(4)

This section will focus on the generic EM adequacy in evaluating sodium chemical reactions, RN release/transport and functional containment analyses. Each mechanism will be discussed separately. Although not required, guidance specified in RG 1.203 [1] will be utilized to assist in performing the closure relation assessment for the generic adequacy of the Source Term EM. RG 1.203 will serve as guidance for evaluating integration and establishing potential processes for adequacy of the EM in relationship to the source term scenarios.

5.1 Closure Relations (Bottom-Up)

RG 1.203 describes a process to determine the EM pedigree and applicability to simulate physical processes. With respect to the four codes identified above, part of the initial review for this process was conducted. The physical phenomena and characteristics that make up a source term and its interaction with systems have been evaluated, and the four codes noted above are dispositioned on their capabilities to be able to model these physical processes. Further information is provided below with respect to a summary of each code's capability related to assessing physical processes.

Closure relations applicability:

1. [[ ]](a)(4) – Discussions related to closure relations are discussed in Section 4.2.1.4. [[ ]](a)(4)
2. [[ ]](a)(4) – Discussions related to closure relations are discussed in Section 4.2.2.4. [[ ]](a)(4)
3. [[ ]](a)(4) – Discussions related to closure relations are discussed in Section 4.2.3.4.
4. RADTRAD – Discussions related to closure relations are found in Section 4.2.4.4.

5.1.1 Determine Closure Model Pedigree and Applicability

This section addresses EMDAP Step 13. The pedigree evaluation relates to the physical basis of a closure model, assumptions and limitations attributed to the model, and details of the adequacy characterization at the time the model was developed. The applicability evaluation relates to whether the model, as implemented in the code, is consistent with its pedigree or whether use over a broader range of conditions is justified.

It is interpreted that pedigree relevant to software relates to the lineage, origin and history of the software. The lineage and ancestry of each software code that is part of the Source Term EM is discussed below. The software codes previously described all have been previously assessed by the code developers with respect to benchmarking efforts, validation efforts (through comparison data). The paragraphs below also address other factors related to considerations for determining an overarching model pedigree and how it integrates with a methodology through use of multiple software codes.

In addition to modeling capabilities, the code selection process and evaluation of “pedigree” is governed by several other key factors. The following metrics are used to assess the suitability of the codes for evaluating the transient performance of the proposed Natrium design for quantified events with fuel failure and potential subsequent RN releases. The Source Term EM structure included addressing normal operations, and EPZ applications. Hence, modeling of physical processes is not limited merely to events that have the potential for a fuel failure.

With respect to the individual code pedigrees related to modeling physical processes, several factors need to be addressed. The information of the Source Term EM pedigree and applicability is given in accordance with the Source Term model development plan. Other factors that can address a software code pedigree can include the following attributes:

1. Development Status: Current state of the code development (Active or Inactive) and maintenance

2. Code Availability: Availability of the code and the perceived ease of obtaining either, or a combination of, the executable or source code to be used to support the Natrium design performance evaluation within a limited timeframe.

Table 5-2 highlights the code evaluation process. The code evaluation summarized several factors relevant to the EM adequacy, which is further discussed below.

Table 5-2. Code Evaluation

Code Trait	[[		]](a)(4)	RADTRAD
Development Status	Active	Active	Active	V3.10 Not Active
Code Availability	Accessible	Accessible	Accessible	Accessible

Another factor is the familiarity of the regulatory body with utilization of the Source Term EM specified codes, regarding “pedigree” whether in advanced nuclear or LWR submittals. The lineage (pedigree) of each code and how it has been applied in other SR source term submittals is relevant for the code’s adequacy within an integrated model and discussed next in generalities.

1. [[

]](a)(4)

[[

]]$^{(a)(4)}$

4. RADTRAD 3.10 - The RADionuclide, Transport, Removal, and Dose Estimation (RADTRAD) code is a licensing analysis code used to show compliance with nuclear plant siting criteria for the off-site radiation doses at the EAB and the LPZ and to assess the occupational radiation doses in the control room (CR) and /or Emergency Offsite Facility for various LOCAs and non-LOCA DBAs. [20] RADTRAD uses a combination of tables and numerical models of source term reduction phenomena to determine the time-dependent dose at user-specified locations for a given accident scenario. It also provides the inventory, decay chain, and dose conversion factor tables needed for the dose calculation. The RADTRAD code can be used to assess occupational radiation exposure, typically in the control room, as well as off-site doses, and to estimate the dose attenuation due to modification of a facility or accident sequence. The AST analysis has been endorsed for applicant usage by the NRC through SECY 98-154, while Regulatory Guide 1.183 details the revised standard review plan. RADTRAD was developed for the USNRC Division of Reactor Program Management/Office of Nuclear Reactor Regulation by SNL and ITSC. NUREG/CR-6604 (1998) documents the code development for the USNRC. RADTRAD has been used extensively in LWR applications, and footnote 15 of Regulatory Guide 1.183 indicates that this code incorporates suitable methodologies for evaluating radiological source terms for DBAs. RADTRAD was upgraded to RADTRAD 3.10 by the same original code developer (ITSC now operating as Serco-NA) and thus, the lineage for its applications and adequacy has been carried forward based upon extensive testing, and subsequent utilization in the industry.

5.1.2 Prepare Input and Perform Calculations to Assess Model Fidelity and/or Accuracy

This section addresses EMDAP Step 14. It addresses performance of calculations to assess the EM fidelity or adequacy, and scalability. Evaluation of the EM fidelity and scalability is instrumental in understanding whether a model is adequate or not. Likewise, it can identify gaps in a model that potentially could require correction prior to implementation. Future assessments will be specifically addressed in the Source Term EM to assess individual model fidelity, accuracy, and scaling for source terms. This assessment will address model fidelity/accuracy and is currently being projected to start in the first part of 2024.

5.1.3 Assess Scalability of Models

This section addresses EMDAP Step 15. The scalability evaluation is limited to whether the specific model or correlation is appropriate for application to the configuration and conditions of the Natrium plant and transient under evaluation.

As noted in the previous section a future integrated model evaluation assessment is scheduled to start in the first part of 2024 relevant to model fidelity and scalability for the Source Term EM. Relevant for this Topical Report explicit details related to scalability of the models for the Source Term EM are not provided herein but will be provided later.

5.2 Integrated Evaluation Model (Top-down)

This section will address the integrated approach of the EM, by first evaluating each software code with respect to model adequacy. This includes the evaluation of each code from an integrated approach, per steps 16 through 19 of the EMDAP. Figure 4-1 shown in a previous section illustrates the interdependence of each code in relationship to the potential release categories.

5.2.1 Determine Capability of Field Equations and Numeric Solutions to Represent Processes and Phenomena

This section addresses EMDAP Step 16. Each code has been evaluated with respect to the field equations, numeric solutions and its applicability for simulating system components. Thus, each code will be further discussed with respect to the field equations and numeric solutions:

[[

]](a)(4)

RADTRAD - field equations, numeric solutions and applicability for simulating releases with dose predications have been previously discussed in Section 4.2.4 for RADTRAD.

5.2.2 Determine Applicability of Evaluation Model to Simulate System Components

This section addresses EMDAP Step 17. It addresses the integrated approach of the EM, by first evaluating each software code with respect to model adequacy. The second step as part of this process includes evaluating each code for the capability to simulate system components and then evaluating that capability from an integrated approach.

[[

]](a)(4)

RADTRAD - See section 4.2.4 for a full discussion of RADTRAD capabilities to simulate processes and ability to evaluate potential releases from an integrated approach.

When the codes are integrated with an application, they form a framework that both addresses normal and plausible accident event scenarios with respect to potential radiological releases and consequences from such.

5.2.2.1 Prepare Input and Perform Calculations to Assess System Interactions and Global Capability

This section addresses EMDAP Step 18. The fidelity evaluation compares EM calculated data with measured data from component and integral tests and, where possible, plant transient data.

The EM evaluation for integrated code usage is an ongoing process with respect to system interactions and global capability. However, several key elements have already been documented regarding code verifications, PIRT process, code validations and gap assessments which are all part of assessing system interactions and the global capability for the source term methodology. With respect to code verifications four reports were generated and are discussed below.

5.2.2.2 Code Verifications

Code verifications have been conducted for [[ ]](a)(4) as part of the Software Quality Assurance Program. RADTRAD 3.10 code verification has been previously conducted via the code vendor as part of the vendor's NQA-1 program. [[ ]](a)(4) code verification has been previously conducted via the code vendor as part of the vendor's NQA-1 program. Both [[ ]](a)(4) and RADTRAD are developed under ASME NQA-1 programs and the code verification tests are covered by the code developers. TerraPower is also conducting an internal verification of the codes described in this section. The technical, quality, and documentation requirements for the software codes noted above are discussed within the documents referenced, any assumptions and open items relevant to the codes are tracked within TerraPower. A review of assumptions relevant to the noted codes as part of the final evaluations has not identified any significant barrier to utilization of these codes for the Source Term EM. The formal documented technical evaluations for the four codes noted above also included development and identification of critical characteristics.

5.2.2.3 PIRT

The purpose of this task was to develop a credible PIRT to support radiological source term analysis based on the judgment of internal and external subject matter experts in a panel discussion. A PIRT panel was conducted to discuss the initial PIRT and collect the expert opinions necessary to finalize the PIRT. The final PIRT was used to determine the requirements for physical model development, scalability, validation, and sensitivities studies to assess the significance and state of knowledge of phenomena and processes that may affect the Natrium source term considering a representative group of transient scenarios. The PIRT was completed prior to building and assessing the EM. Section 2.4 of this topical report addresses in detail the PIRT process and results for source term consideration.

Related to some of the PIRT identified phenomena the following information is provided that was obtained after the initial report. Included in the assessment for validity of the release fractions and rates to be potentially utilized within the proposed source term framework is a comparison to a recent report issued by the International Atomic Energy Agency (IAEA) [27]. The IAEA technical report addressed simulations for generating release fractions of RNs to the cover gas from vapor fractions for fuel operating in an SFR. This work was the final recent report of coordinated international research for SFRs under hypothetical severe accident conditions.

The accident sequence considered for the IAEA work was an Unprotected Loss of Flow Accident (ULOFA) and therefore, is relevant to this topical report. Table 53 of this report identified a potential high-risk phenomenon for a ULOF+ and one of those phenomena is release rate and fuel migration for RNs. The work documented in [27] describes generation of not only release fractions for no mixing conditions, but also ideal and real mixing conditions, at various fuel temperatures. Even though the work reported in [27] was analyzed for uranium oxide fuel types, comparisons and discussions were also provided in relationship to metal fuels [28]. Table 47 of [27] illustrated a very conservative no mixing case at high temperature (1156 K) that the RF for all forms of Iodine would be 0.435. This can be compared against real mixture simulations, where the predicted Iodine RF would be lower at 4.82E-06 for the same temperature.

This report also generated volatile fission produce releases from melted fuel. It was concluded that the release from melted fuel lasts less than one minute for volatile fission products. For non-volatile fission products the release lasts for about 10 minutes, with exception to Sr which has a release rate of 20 hours. For consideration in this framework is the work documented with metal fuels which identified that metallic Sr is more volatile than its oxide forms [28]. The work documented by Schram also noted this observation with Eu (volatility) and speculated that it was due to less oxygen being available for reaction. Hence, the potential volatility of both Sr and Eu need to be considered with respect to release fraction calculations in the Natrium source term methodology. With respect to the fission products released from the liquid sodium itself to the cover gas this can be used as a comparison benchmark for the source term methodology tasks.

For some RN species it was reported that the activity for some fission products can take a significantly longer length of time before being depleted from the sodium. Table 26 of [27] reports that there is essentially very little Cs, Te, and I in the cover gas based upon the research simulations. [[ ]]$^{(a)(4)}$ Since it was reported that Sr and Eu could be more volatile this may need to be addressed as part of the source term methodology. Thus, with the proposed methodology framework there is some recent international research to be utilized for comparison and validation purposes.

Code Validations

Code validations are ongoing for [[ ]]$^{(a)(4)}$ with respect to utilization with the Source Term EM. With respect to [[ ]]$^{(a)(4)}$ there is a substantial suite of benchmark analyses presented within the standard vendor-provided [[ ]]$^{(a)(4)}$ code qualification package report supporting the code's generic NQA-1 status, which provides evidence demonstrating [[ ]]$^{(a)(4)}$ capabilities to analyze a broad range of situations with reasonable accuracy by demonstrating acceptable agreement with testing or operational data used in the qualification benchmarks. As such, the benchmark analyses provided with the code [29] demonstrate that, when appropriately modeled, [[ ]]$^{(a)(4)}$ is generally capable of modeling the phenomena of interest relevant to Source Term EM and providing reasonable predictions of the parameters quantified with the relevant FOMs.

[[

]](a)(4) Additionally, RADTRAD validation is generically discussed in NUREG/CR-6604.

5.2.2.4 Model Acceptance Assessment

The detailed EM acceptance assessment is planned to begin in the first part of 2024. This section of the topical report will be updated once that information becomes available. Acceptance test plans for each individual code mentioned have been formally completed.

5.2.2.5 Strategy for Addressing Gaps

Current gaps that have been identified relevant to the Source Term EM are discussed in this section. Further updates to this topical may be made as resolution of each identified gap is completed.

[[

]](a)(4)

RADTRAD Gaps – there are no known gaps documented relevant to RADTRAD regarding the Source Term EM.

5.2.3 Assess Scalability of Integrated Calculations and Data for Distortions

This section addresses EMDAP Step 19. Future assessments will be specifically addressed in the Source Term EM assessment for individual model fidelity, accuracy, and scaling for sources. This task which will address model fidelity and accuracy is currently being projected to start the first part of next year. As part of this task, it is anticipated that it will address the integrated calculations and consideration for data distortions.

5.3 Determine Evaluation Model Biases and Uncertainties

This section addresses EMDAP Step 20. As part of the EM process biases and uncertainties will be addressed for all LBEs with exception to the DBAs identified in Section 6.2. The DBAs identified as part of the EM will use an approach that considers conservatisms. The Source Term EM will address prediction of FOMs through incorporation of biases and uncertainties into the various code mathematical models. When addressing the Source Term EM uncertainties, it is anticipated to involve three steps. First, the sources of uncertainty need to be characterized. Second, the propagation of uncertainties through the various codes utilized for the source term need to be quantified. Lastly, the Source Term EM output needs to include sensitivity analysis of the output. The four software codes to be utilized for the Source Term EM each need to be evaluated on their own related to characterizing sources of uncertainties and propagation of uncertainties. Also, the inter-relation of the codes with respect to propagation of uncertainties is yet to be evaluated. For example, currently it is anticipated that output from [[ ]](a)(4) may need to feed into [[ ]](a)(4) How uncertainties propagate through this step of the model will need to be addressed.

The characterization of uncertainties is anticipated to consist of evaluating whether the uncertainty is epistemic or aleatory. In this context aleatory uncertainty stems from inherent randomness and is also known as stochastic uncertainty, while epistemic uncertainty stems from lack of knowledge. Epistemic uncertainties could potentially be evaluated through probability density functions or a cumulative distribution function. Aleatory uncertainties could potentially be evaluated using inter valued quantities. Likewise, combinations of epistemic and aleatory uncertainty can be used. The second step involves propagation of the uncertainty and statistical sampling procedures such as Monte Carlo sampling or Latin Hypercube sampling of uncertain quantities with repeated execution of the model could be introduced. Each type of uncertainty with each code may be treated and evaluated separately. Individually the type of uncertainty, whether epistemic or aleatory, may be treated separately. Samples from the epistemic uncertainty values could produce possible realizations of the FOM values, whereas samples from the aleatory uncertainty values produce aleatory uncertainty in the FOM values. The third step of uncertainty propagation involves analysis of the Source Term EM output through interpretation of a collection of calculations for each software code. Propagation of uncertainties will be addressed using guidance as noted in Reference [30].

Some sources of model prediction biases and uncertainties can be a result of the following:

1. Inadequacy of equation forms
2. Inadequacy, incorrect, or improper application of closure correlations
3. Limitation of numerical techniques and methods
4. Uncertainty in specifying initial and boundary conditions
5. Biases and uncertainties in validation data

These sources will be evaluated as part of the overall uncertainty assessment.

Additionally, a methodology for developing uncertainty treatment of high-risk phenomena is being drafted for the Natrium project. This overarching methodology for uncertainty treatment has been initiated and will be ongoing. Updates to this topical will be provided when the uncertainty treatment has been completed.

Any activity to be utilized for the source term uncertainty application is dependent also on whether a rigorous approach is warranted or not. For example, if a sufficient margin is present for source term evaluations, use of a bounding uncertainty can simplify the process. This latter approach may be the considered mitigation for some of the high-risk phenomena identified in Table 2-7. The documented analysis of the Source Term EM related to code outputs and their sensitivities is anticipated to justify the level of rigor required.

In summary, from preliminary observations and reviews both the integrated approach for Natrium Source Terms and code selections reviewed have been found to illustrate a process for assessing and justifying the adequacy of the proposed EM.

6 NATRIUM SAMPLE ANALYSIS RESULTS

At the time of this writing, the majority of source term analyses have not been performed in sufficient detail to warrant inclusion in this report. However, there has been some relevant work performed as part of larger scoping efforts. Details concerning two sample calculations are contained in Appendix A.

7 ADEQUACY DECISION

The adequacy decision is the culmination of the adequacy demonstration process. Questions concerning the adequacy of the EM will be addressed throughout the entire EMDAP. At the end of the process, the adequacy will be questioned again to ensure that all earlier answers are satisfactory and that intervening activities have not invalidated previous acceptable responses. If unacceptable responses indicate significant EM inadequacies, the code deficiency will be corrected and the appropriate steps in the EMDAP will be repeated to evaluate the correction.

This will be the last task to be performed and documented prior to submitting a final update of the source term evaluation methodology in support of the Natrium application submittal to the NRC.

8 CONCLUSIONS AND LIMITATIONS

8.1 Conclusions

TerraPower is requesting NRC approval of the Source Term EM methodology documented in this report for use by future applicants utilizing the Natrium design as an appropriate and adequate means to calculate radiological source terms in evaluation of the radiological consequences of quantified events (as described in Section 1.5). This approval is subject to the limitations described below.

8.2 Limitations

This section describes the limitations of radiological source term methodology presented in this report. Each limitation must be addressed in safety analysis reports associated with licensing application submittals which use this methodology, or justification provided for why the limitation may remain open.

1. The methodology is limited to a Natrium design that has a pool-type, SFR design with metal fuel and sodium bond as described in Sections 1.3 and 2.3.1. Changes from these design features will be identified and justified in Safety Analysis Reports of Natrium license applications.

2. The fuel failure fractions during normal operation and transient conditions are subject to the qualification of Type 1 fuel.

3. If bonded sodium is not utilized in subsequent fuel designs, additional information shall be provided to justify the fission product release behavior from metal fuel to the gas plenum.

4. The sodium pool scrubbing and associated RN retention within the primary sodium coolant is limited to where the bulk sodium is in subcooled conditions.

5. The methodology will be used to determine the RN inventory up to the last barrier prior to the environment and provide the inputs to subsequent radiological consequence analyses. The radiological consequence analysis methodology will determine the on-site and off-site radiological consequences.

6. Adequate verification and validation assessment information should be made available to the NRC staff as part of future submittals supporting the codes that make up the evaluation model. This verification and validation information should be justified to reasonably bound the operational envelope for the design for any applicant referencing the Source Term EM methodology.

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APPENDIX A

Sample Calculations

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Table A-1. [[ ]](a)(4)

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Table A-2. [[ ]](a)(4)

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Results

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Table A-3. [[ ]](a)(4)

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A.2 [[

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Table A-4. [[ ]][(a)(4)]

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Table A-5. [[ ]](a)(4)

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