September 29, 2023

TP-LIC-LET-0098
Project Number 99902100

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

Subject: Transmittal of TerraPower, LLC Topical Report, "Design Basis Accident Methodology for In-Vessel Events without Radiological Release," Revision 0

This letter transmits the TerraPower, LLC (TerraPower) Topical Report "Design Basis Accident Methodology for In-Vessel Events without Radiological Release," Revision 0 (enclosed). The report contains an overview and description of the model developed to evaluate in-vessel Design Basis Accident events for the Natrium™ Plant[1].

TerraPower requests the NRC's review and approval of the 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 [[ ]]$^{(a)(4), ECI}$.

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

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

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

Sincerely,

Ryan Sprengel

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, "Design Basis Accident Methodology for In-Vessel Events without Radiological Release," Revision 0 – Non-Proprietary (Public)
3. TerraPower, LLC Topical Report, "Design Basis Accident Methodology for In-Vessel Events without Radiological Release," 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: September 29, 2023

George Wilson

George Wilson
Vice President, Regulatory Affairs
TerraPower, LLC

ENCLOSURE 2

TerraPower, LLC Topical Report
"Design Basis Accident Methodology for In-Vessel Events without Radiological Release" Revision 0

Non-Proprietary (Public)

Controlled Document - Verify Current Revision

TOPICAL REPORT

Document Number:	TP-LIC-RPT-0004	**Revision:**	0
Document Title:	Design Basis Accident Methodology for In-Vessel Events without Radiological Release		
Functional Area:	Licensing	**Engineering Discipline:**	Safety & Licensing
Effective Date:	9/29/2023	**Released Date:**	9/29/2023
		Page:	1 of 85

Approval

Title	Name	Signature	Date
Originator, Licensing Engineer	Matthew Presson	Electronically Signed in Agile	9/29/2023
Reviewer, Licensing Manager	Nick Kellenberger	Electronically Signed in Agile	9/29/2023
Approver, Director of Licensing	Ryan Sprengel	Electronically Signed in Agile	9/29/2023

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

REVISION HISTORY

Revision No.	Effective Date	Affected Section(s)	Description of Change(s)
0	09/29/2023	All	Initial Release

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EXECUTIVE SUMMARY

This report summarizes the approach taken to satisfy the guidance outlined in Regulatory Guide (RG) 1.203 Evaluation Model Development and Assessment Process (EMDAP) for in-vessel design basis accident (DBA) events without radiological release in the Natrium™ reactor, a TerraPower & GE-Hitachi Technology.

Within RG 1.203, six basic principles are identified as important to follow in the process of developing an EM. The first four principles deal with the EMDAP itself, whereas the remaining two principles focus on the quality assurance protocol and the need for comprehensive, accurate, and up-to-date documentation. The content of this topical report is summarized by briefly outlining the approach for satisfying the requirements of the four principles that deal with EMDAP in the subsequent paragraphs.

An adequacy decision is achieved when Regulatory Positions 1.1 through 1.4 have been satisfactorily addressed (see RG 1.203, Section 1.5, p. 20). Regulatory Positions 1.1 through 1.4 are discussed below from the perspective of achieving this objective.

It is noted that the strategy to follow the EMDAP defined in RG 1.203 is still under development for the DBA methodology for in-vessel events without radiological release. The objective is to develop a conservative methodology to determine the Natrium EM adequacy; therefore, an uncertainty methodology will not be discussed in this report. Instead, the conservative methodology which is still under development will be shown to be suitably conservative.

Regulatory Position 1.1 (RP 1.1) concerns the determination of the requirements for the EM. RP 1.1 is addressed in EMDAP Element 1. Key events and scenarios relevant to in-vessel DBAs without radiological release were identified together with the Figures of Merit (FOMs) and phenomena which are highly-ranked in importance in the five Phenomena Identification and Ranking Table (PIRT) studies that were conducted (see Chapter 2). [[ ]](a)(4) highly-ranked phenomena were identified and qualitatively described. Based on historically successful sodium-cooled fast reactor (SFR) experiments, design and deployment of measurement diagnostics, and subsequent analyses, successful mathematical modeling methods have been identified. The SAS4A/SASSYS-1 (SAS) code serves as the basis for the EM.

RP 1.2 focuses on the development of an assessment base consistent with the determined requirements. RP 1.2 is addressed in EMDAP Element 2 (Chapter 3). The assessment base forms the content of the Natrium EM code assessment matrix. The required data sets that will characterize the highly-ranked phenomena and populate the Natrium code assessment matrix will be obtained from:

- [[

]](a)(4)

Data from the above planned and vintage experimental facilities are presently included in the preliminary EM code assessment matrix.

Regulatory Position 1.3 (RP 1.3) concerns the development of the EM. RP 1.3 is addressed in EMDAP Element 3 (Chapter 4). The EM is based on the SAS code that was developed and is being maintained and revised to accommodate the needs of this EM development effort by Argonne National Laboratory (ANL). The ingredients that distinguish the Natrium EM from earlier versions of the SAS are:

- [[

]](a)(4)

Regulatory Position 1.4 (RP 1.4) concerns the assessment of the EM adequacy: RP 1.4 is addressed in Element 4 (Chapter 5). The calculations and evaluations performed demonstrate the closure relationships and the integrated EM are satisfactory. The calculations will demonstrate that the analyses are “suitably conservative” and thus demonstrate that the Natrium EM is adequate. The selection of conservative assumptions will be informed by the quantitative uncertainty analysis of consequences that will be performed for the corresponding Design Basis Event (DBE) – consistent with the NEI 18-04 methodology.

This report documents the Natrium In-Vessel DBA EM adequacy. Certain aspects of the EM adequacy demonstration remain in development and are noted throughout the report. It is acknowledged that this report does not contain the complete technical basis that would be expected in a full transient and safety analysis methodology report. Several sections describe actions that are planned to be taken by TerraPower, and information generated by these actions will be provided through revisions to this report, supplemental documents, or future engagements prior to the method's use with an operating license.

TerraPower requests NRC review and approval of the proposed DBA methodology documented in this report for use by future applicants utilizing the Natrium design as an appropriate and adequate means to evaluate in-vessel DBA events without radiological release.

ACRONYMS

Acronym	Definition
AHX	Sodium-Air Heat Exchanger
ANL	Argonne National Laboratory
AOO	Anticipated Operational Occurrences
BDBE	Beyond Design Basis Events
BOL	Beginning of Life
BOP	Balance of Plant
CDF	Cumulative Damage Fraction
CGD	Commercial Grade Dedication
CPA	Construction Permit Application
CRD	Control Rod Drive System
DBA	Design Basis Accident
DBE	Design Basis Event
DID	Defense-in-Depth
DOE	Department of Energy
EBR	Experimental Breeder Reactor
EM	Evaluation Model
EMDAP	Evaluation Model Development and Assessment Process
EOC	End of Cycle
F-C	Frequency-Consequence
FDE	Finite Difference Equation
FFTF	Fast Flux Test Facility
FOM	Figure of Merit
FSAR	Final Safety Analysis Report
GEH	GE-Hitachi
GV	Guard Vessel
HCF	Hot Channel Factor
HPR	Hot Pin Ratio
IAC	Intermediate Air Cooling
IET	Integral Effects Test
IHT	Intermediate Heat Transport System
IHX	Intermediate Heat Exchanger
IRACS	Intermediate Reactor Auxiliary Cooling System
ISP	Intermediate Sodium Pump
IVS	In-Vessel Storage
IVTM	In-Vessel Transfer Machine
KAERI	Korea Atomic Energy Research Institute
LBE	Licensing Basis Event
LMR	Liquid Metal Reactor

Acronym	Definition
LOFWOS	Loss of Flow Without Scram
LOHS	Loss of Heat Sink
LOOP	Loss of Offsite Power
LWR	Light Water Reactor
NEI	Nuclear Energy Institute
NQA	Nuclear Quality Assurance
NRC	Nuclear Regulatory Commission
NSRST	Non-Safety-Related with Special Treatment
NSS	Nuclear Island Salt System
NST	Non-Safety-Related with No Special Treatment
PCT	Peak Cladding Temperature
PDE	Partial Differential Equation
PHT	Primary Heat Transport System
PIRT	Phenomena Identification and Ranking Table
PNC	Power Reactor and Nuclear Fuel Development Corporation
PRISM	Power Reactor Innovative Small Module
PSAR	Preliminary Safety Analysis Report
PSP	Primary Sodium Pump
RAC	Reactor Air Cooling
RAC	Reactor Vessel Air Cooling
RCC	Reactor Core and Core Components System
RES	Reactor Enclosure System
RG	Regulatory Guide
RP	Regulatory Position
RSF	Required Safety Functions
RV	Reactor Vessel
RVACS	Reactor Vessel Auxiliary Cooling System
RVH	Reactor Vessel Head
RWAP	Rod Withdrawal at Power
RXB	Reactor Building
SAS	SAS4/SASSYS-1
SET	Separate Effects Test
SFR	Sodium-Cooled Fast Reactor
SHRT	Shutdown Heat Removal Test
SHX	Sodium-Salt Heat Exchangers
SR	Safety-Related
SRP	Standard Review Plan
SSC	Structures, Systems, and Components
TATNF	Time-at-Temperature No-Failure

Acronym	Definition
TWR	Travelling Wave Reactor

1 INTRODUCTION

This report documents the evaluation method developed for in-vessel DBA events without radiological release which are associated with the Natrium™ Reactor Plant, the EM development process, the resulting EM, and identifies items which require additional development. Certain aspects of the adequacy demonstration for the EM remain in development and are noted throughout the report. Overarching TerraPower methodology development guidance and RG 1.203, *Transient and Accident Analysis Methods* [3] were used to guide the EM development process.

1.1 Regulatory Requirements and Guidance

DBA postulated accidents "… are used to set design criteria and limits for the design and sizing of safety-related systems and components." per the Standard Review Plan (SRP) (NUREG-0800) 15.0[1]. Further, as noted in NUREG-2122: A DBA "…is a postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structure, and components necessary to ensure public health and safety." The definition put forth in Nuclear Energy Institute (NEI) 18-04 is: [2]

> "Postulated event sequences are used to set design criteria and performance objectives for the design of Safety Related SSC. DBAs are derived from DBEs based on the capabilities and reliabilities of Safety-Related SSCs needed to mitigate and prevent event sequences, respectively. DBAs are derived from the DBEs by prescriptively assuming that only Safety Related SSCs are available to mitigate postulated event sequence consequences to within the 10 CFR 50.34 dose limits."

As shown in Figure 1-1, DBAs are derived from DBEs and have no frequency assigned. The DBAs meet the definition given in NEI 18-04 and were obtained using the NEI 18-04 processes as noted in the next paragraph.

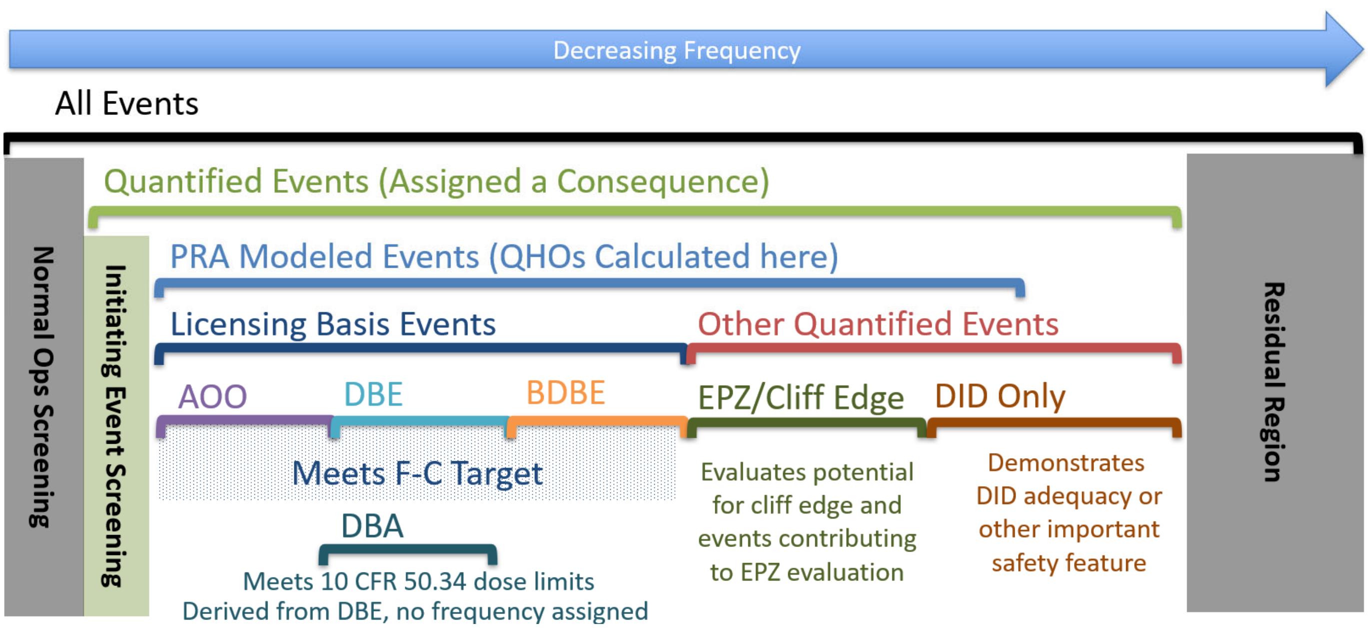

Figure 1-1. Event Type Line Diagram by Frequency.

[1] See NEI 18-04, Table 3-1, p. 6.

NEI 18-04 & 21-07: Methodologies developed to identify the postulated events associated with in-vessel events without radiological release have been performed to define the DBAs considered in this report. These methodologies conform to the "…technology-inclusive, risk-informed, and performance-based process for the selection of Licensing Basis Events (LBEs); safety classification of SSCs and associated risk-informed special treatments; and determination of Defense-in-Depth (DID) adequacy for non-light water reactors." [2] The processes described in NEI 18-04 are "…acceptable processes for selection of LBEs; safety classification of SSCs and associated risk-informed special treatments; and determination of DID adequacy applicable to a technology-inclusive array of advanced non-Light Water Reactor (LWR) designs." [2] By following the guidance in [2] and NEI 21-07: Technology Inclusive Guidance for Non-LWRs: Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology [4], TerraPower has developed the basis for evaluations that can be used to demonstrate compliance with 10 Code of Federal Regulations (CFR) 50.34 for both a Preliminary Safety Analysis Report (PSAR) for a Construction Permit application and the Final Safety Analysis Report (FSAR) for an Operating License application.

Content of This Report: The following topics are addressed:

- DBA events selected for analysis (Chapter 1.4),
- Required EM capabilities for performing in-vessel DBAs without radiological release (Chapter 2)
- EM assessment base development for the selected DBAs (Chapter 3)
- EM development for the analysis of the selected DBA events (Chapter 4)
- Bottom-up and Top-down EM adequacy assessment for the DBA events (Chapter 5)
- Sample analysis results (Chapter 6)
- EM adequacy decision (Chapter 7)

The report structure given above describes how the EMDAP methodology has been applied to the development, assessment, and the determination of adequacy of the EMs used to analyze the DBAs for in-vessel events without radiological release.

The Evaluation Model Concept: as defined in RG 1.203[1], "…establishes the basis for methods used to analyze a particular event or class of events. This concept is described in 10 CFR 50.46 for loss-of-coolant analyses but can be generalized to all analyzed events described in the SRP." As such:

> "An evaluation model (EM) is the calculational framework for evaluating the behavior of the reactor systems during a postulated transient or design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event, as illustrated by the following examples:
>
> 1. procedures for treating the input and output information (particularly the code input arising from the plant geometry and the assumed plant state at transient initiation)
> 2. specification of those portions of the analyses not included in the computer programs for which alternative approaches are used
> 3. all other information needed to specify the calculational procedure.

[1] See Section B, Discussion, p. 3.

> The entirety of an EM ultimately determines whether the results are in compliance with applicable regulations. Therefore, the development, assessment, and review processes must consider the entire EM.
>
> The reader should note that this regulatory guide also uses the term "model," which should be distinguished from the evaluation model or EM. In contrast to the EM as defined here, "model" (without the "evaluation" modifier) is used in the more traditional sense to describe a representation of a particular physical phenomenon within a computer code or procedure."

The EM used to evaluate postulated in-vessel DBAs without radiological release is centered on the use of the SAS code.

The adequacy of the EM is achieved by following the EMDAP as shown in flow chart form in RG 1.203, Figure 1[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 (Chapter 2)
Element 2	Develop assessment base (Chapter 3)
Element 3	EM development (Chapter 4)
Element 4	EM adequacy assessment (Chapter 5)

Element 1 focuses on establishing the exact application envelope for the EM and identifying the importance of constituent phenomena, processes, and key parameters within that envelope. Chapter 2 documents the determination of: (i) the necessary capabilities of the EM by identifying the physics that should be contained in the EM for the transient scenarios, (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 obtain an adequate assemblage of experimental data for use as the 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 contains the central activities of (i) establishing the EM development plan and (ii) constructing the EM. The action of creating the EM development plan (identified as Step 10 in EMDAP) is the central key activity of EMDAP. Within the EM development plan are 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) the documentation of the software design and implementation, (d) the source code verification test report, (e) validation testing report, and (f) the installation package and program upgrade documentation. Therefore, these sections and their associated documentation contain the descriptions of phenomena that must be contained within the EM, the 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 EMs are capable of calculating all the key phenomena within all the nuclear reactor components and within the system as a whole for all the transients listed in Element 1. Based on these parameters, the plan will develop the

[1] See Section B, Discussion, p. 6.

specifications of the experiments and their required measurement uncertainties, the acceptable distortion levels of the experiments to be used to generate validation data, the scale-up of experimental data recorded in experimental facilities much smaller than the full-sized plant, the validation metrics, and the limits within which the determination of EM adequacy. 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 direct 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 and performance of calculations to enable validation studies to be performed through model scalability, and (ii) via top-down considerations, i.e., ultimately demonstration of the 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 requirements of the EM development plan enables the plant event analyses to be performed for licensing purposes.

1.2 Sample Plant Description

The Natrium reactor is an 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 conceptual 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 Nuclear Island and equipment supporting energy production 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 SR SSCs needed to achieve safety goals.

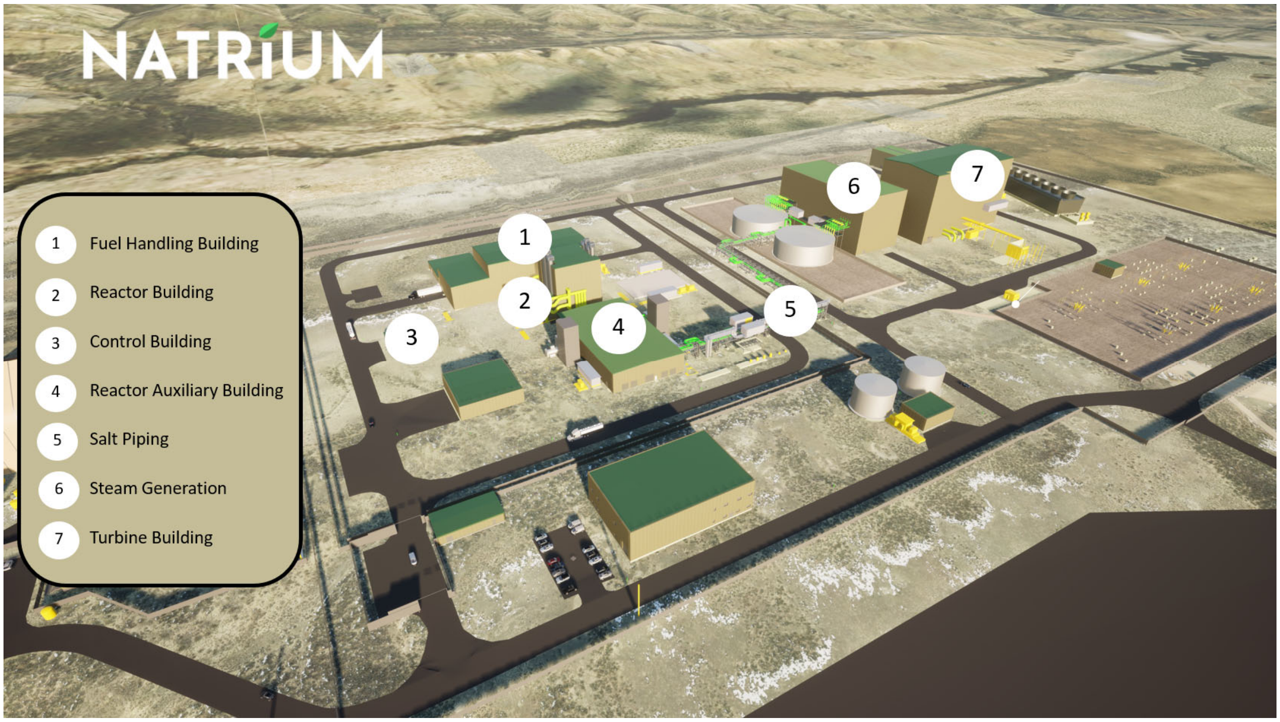

Figure 1-2. Plant Layout

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 penetrations through the sides of the Reactor Vessel (RV), thereby eliminating loss of coolant accidents involving primary pumps and piping. The primary sodium pool operates 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 Sodium-Salt Heat Exchangers (SHXs). These SHXs in the Nuclear Island heat salt received from the cold salt tank(s) in the Energy Island. The heated salt is then returned to the Energy Island for storage in the hot salt tank(s), 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 reactor operates at a thermal power of 840 MW while the plant produces 336 MWe steady-state and 500 MWe peak power.

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

- The NSRST 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 prevent the 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 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, or loss of heat sink.

The high boiling point of sodium allows reactor operation at atmospheric pressure. A close-fitting Guard Vessel (GV) surrounding the RV 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.

The Intermediate Air Cooling (IAC) transfers heat to the atmosphere (the final heat sink) from the Sodium-Air Heat Exchangers (AHXs). Forced flow heat removal via IAC serves as the normal shutdown cooling system for outages. There are two trains, one for each primary heat exchanger. The IAC can also operate in a passive flow mode. Simple operation of a fail-open electromagnetic damper initiates passive cooling. Use of active forced circulation through the IAC supports normal controlled cooling operations (such as during a refueling outage) and provides a 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 RAC system and enable the IAC and its support systems 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. The RAC is always operating and requires no power, people, or control action to perform its function. The RAC heat removal function relies on the natural circulation 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 outer wall via a combination of radiative and convective heat transfer. This air is then returned to atmosphere.

Figure 1-3 provides an illustration of how heat may be removed by both the IAC and RAC.

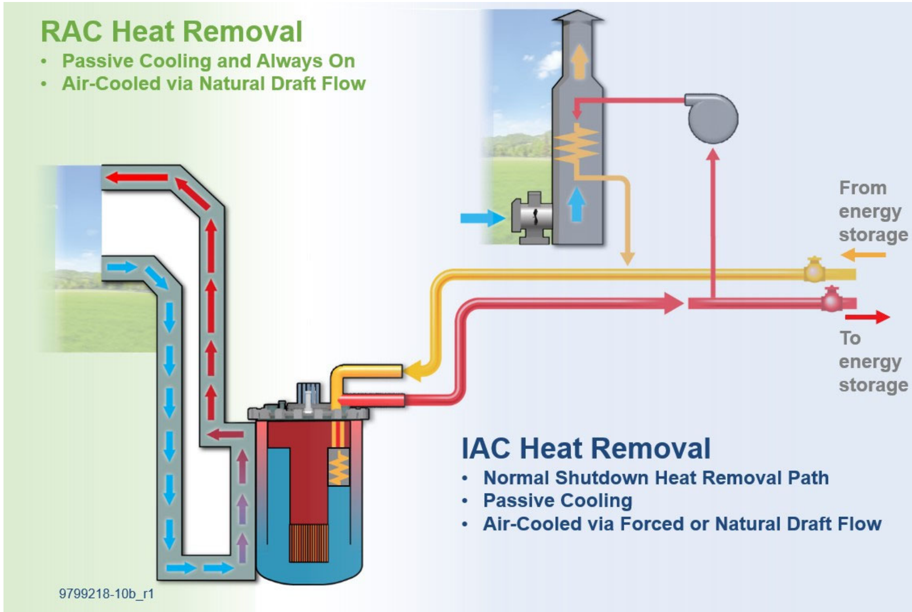

Figure 1-3 IAC and RAC Heat Removal

The Nuclear Island is composed of six major buildings: reactor, fuel handling, control, electrical, reactor auxiliary, and fuel auxiliary buildings. The RXB, see Figure 1-4, 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.

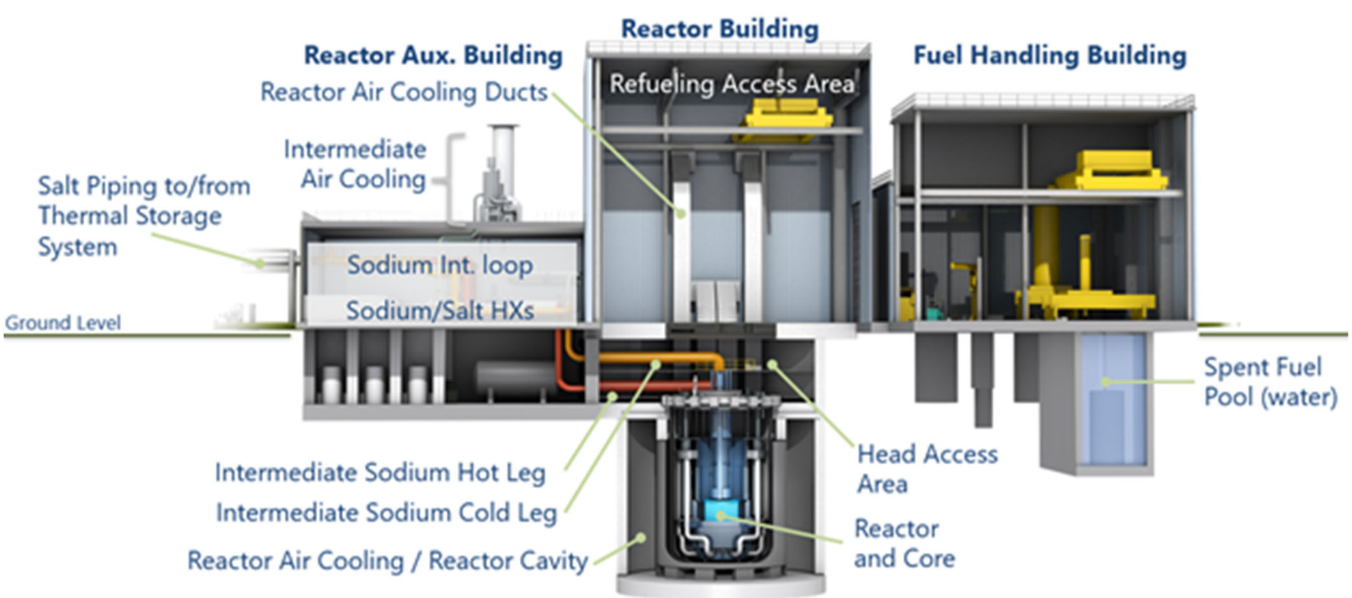

Figure 1-4. Conceptual Elevation View

The Primary Heat Transport System (PHT) is contained within the RV and consists of the IHX, the Primary Sodium Pumps (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 Intermediate Heat Transport System (IHT) sodium coolant transferring heat from the PHT to the Nuclear Island Salt System (NSS), exits the bottom of the IHXs and enters the cold pool. Cold pool sodium flows downward to the PSP inlet plenums which are located 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 the rotatable 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.

1.3 Safety Classification

The Natrium plant uses three safety classification levels: Safety-Related (SR), 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.

Safety-Related

SSCs selected from those that are available to perform the Required Safety Functions (RSFs) to mitigate the consequences of 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 those that are available and relied on to perform RSFs to prevent the frequency of Beyond Design Basis Events (BDBE) with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C target.

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 LBE 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.

Non-Safety-Related with No Special Treatment

All other SSCs (with no special treatment required)

1.4 In-Vessel Design Basis Accidents Without Radiological Release

Accident sequences evaluated within the PRA identify three categories of LBEs: AOOS, DBEs, and BDBEs. 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

DBAs are not categorized by a mean frequency and are instead derived from the DBEs determined above by only crediting safety-related SSCs. The EM presented in this report is used to evaluate DBAs that occur within the reactor vessel, and which do not involve the release of radioactive material.

2 EVALUATION MODEL CAPABILITY REQUIREMENTS: EMDAP ELEMENT 1

A four-step process was undertaken to define the capabilities of the in-vessel DBA 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 a list of important key phenomena (Section 2.4)

The following subsections describe the content of EMDAP Steps 1 through 4. A preliminary evaluation of the highly ranked phenomena is also presented in Section 2.5.

2.1 Analysis Purpose: EMDAP Step 1

The analysis purpose is to demonstrate that the plant operates such that all relevant acceptance criteria are satisfied under normal operational conditions, and continue to be satisfied during in-vessel DBAs without radiological release. The phenomena and processes inherent to the occurrence of in-vessel DBAs without radiological release are identified as inputs to define the physics, models, and calculational capabilities of the EM.

Three scenarios were selected as representative of the potential events included in the in-vessel DBA envelope, and include the Loss of Offsite Power (LOOP), Rod Withdrawal at Power (RWAP), and Loss of Heat Sink (LOHS). These scenarios were reviewed as part of the PIRT development process.

The EM used for these analyses is conservative, and not best-estimate. Therefore, calculational uncertainties are not calculated in the DBAs. However, as noted in Appendix A of RG 1.203 (see p. A-3) the predictions of the EM, or portions thereof, shall be compared with applicable experimental information to the extent practicable.

2.2 Figures-of-Merit: EMDAP Step 2

FOMs give quantitative standards of acceptance with respect to the safety analysis. Adherence to the limits prescribed by the FOMs provides general requirements for maintaining the Natrium reactor in a safe condition during normal operation and during transients and accidents in terms of quantitative fuel and reactor system design limits. Fuel performance-centered acceptance criteria have been established for in-vessel DBA events.

To identify FOMs for EM development associated with DBAs without radiological release, it is helpful to examine event acceptance criteria for its LBEs.

NEI-18-04 provides guidance for selecting LBEs, safety classification of SSCs and associated risk-informed special treatments, and determination of DID adequacy for non-LWRs. NEI-18-04 uses a set of frequency-consequence criteria (referred to as the F-C Target in that report) to select LBEs. The F-C Target values are based on mean event sequence frequency of occurrence per plant-year and radiation exposure limits, respectively.

The F-C Target values provide a general reference to assess events and evaluate safety margins. Fuel performance, especially fuel failure phenomenon, becomes important in deterministic safety analyses that challenge the F-C target. Key parameters (or mechanisms) that can lead to fuel failure are fuel and cladding temperatures and cladding strain. Coolant temperature was considered in the PIRT process for RV integrity.

Natrium Type 1 fuel has a considerable margin to strain limit (even under the conservative analysis conditions involving thinning the initial cladding thickness by 25%, applying a Fuel-Clad Chemical Interaction model which includes uncertainty in the model fit, using 2σ Hot Channel Factor (HCF) temperatures, and including the creep damage model). The large margin to cladding strain limits provides confidence that transient analysis will also meet design limits. The peak cladding temperature (PCT) is used as a surrogate for the cladding strain limit. The fundamental intent of in-vessel DBA without release analysis is intended to show compliance with the above statement.

The parameters (temperatures of coolant, clad, and fuel center) serve as FOMs. The severity of consequences of a DBA can be evaluated by investigating those parameters. The FOMs and their significances are summarized in Table 2-1. Cladding temperature limits are set to prevent fuel pin failure and to maintain a coolable geometry and ensure fuel pin reliability preventing cladding failure by fuel-clad eutectic reaction. The peak cladding temperature limit is applied to the inner cladding wall surface that may be in contact with fuel.

Table 2-1 Figures of Merit for In-Vessel DBAs Without Radiological Release.

Figure of Merit	Descriptions and Significance
Fuel centerline temperature	The fuel centerline temperature must stay below the fuel solidus temperature to avoid fuel damage. Since the fuel solidus temperature is much higher than the fuel-cladding eutectic reaction onset temperature, it is expected that the PCT will be a much more limiting criteria than the fuel temperature.
Coolant temperature	High coolant temperature may cause sodium boiling in the reactor core, which can result in positive reactivity feedback. In addition, this phenomenon can be used to examine the primary boundary integrity. This FOM is tracked, however the acceptance criteria for time-at-temperature no-failure (TATNF) for peak cladding temperature is designed to preclude boiling.
Time-at-temperature for PCT	The design basis approach and limit values of the PCT were evaluated for application to the Natrium design. For mechanical fuel pin cladding failure criteria, the main options include strain, cumulative damage fraction (CDF), stress, and temperature as primary or dependent criteria parameters. The Natrium design basis has adopted response parameters such as strain, wastage, and temperature rather than CDF and stress criteria because they have a historic precedent, are defensible by existing data, are readily analyzed, and can be measured to validate. These attributes allow for monitoring and surveillance that can confirm analysis predictions and assess remaining life of the fuel system. The time-at-temperature no-failure acceptance criteria incorporate cladding wastage and thermal creep criteria in assessing potential failure. [[ ]]$^{(a)(4)}$

Figure of Merit	Descriptions and Significance
	[[ ]](a)(4)

Following development of the above FOMs and the corresponding PIRT review, development of fuel pin cladding temperature criteria for failure analysis proceeded.

The resulting TATNF criteria above were developed after the initial PIRT, however they are consistent with the FOM used within the PIRT. The temperature ranges used for the PIRT have been updated with the addition of time-dependent acceptance criterion for PCT. Evaluation of the time dependent criteria's potential impact on the PIRT must still be performed.

2.3 Systems, Components, Phases, Geometries, Fields, and Processes Modeled: EMDAP Step 3

The hierarchical system decomposition of the Natrium design follows:

- System: Natrium Plant
- Subsystems:
  - Reactor core and core components system
  - Reactor enclosure system
  - Primary heat transport system
  - Intermediate heat transport system
  - Intermediate air cooling system

    - Control rod drive system
    - Reactor air cooling system
- Modules: physical components, including the following
    - Reactor vessel
    - Intermediate heat exchanger
    - Other heat exchangers (e.g., IAC, SHX)
- Constituents:
    - Liquid sodium
    - Air
    - Argon gas
- Phases: Liquid sodium and gases
- Geometrical configurations
    - Liquid sodium flow direction governed by physical structures.
    - Air flowing through the riser of the RAC and IAC
    - Argon gas in stagnant condition above hot pool
- Fields: composed of constituents
    - Mass
    - Momentum
    - Energy
- Transport processes
    - Inter-component transport of constituents
    - Energy transport from:
        - Fuel to liquid sodium
        - Liquid sodium to structures
        - Components (e.g., ISP and PSP) to liquid sodium
        - Structures to surroundings

2.4 Identification and Ranking of Phenomena and Processes: EMDAP Step 4

The final step (Step 4) of Element 1 is the identification and ranking of phenomena and their knowledge states concerning these phenomena—obtained by performing a PIRT for each scenario of interest within a selected event type, e.g., a loss-of-offsite power. PIRTs for DBAs without radiological consequence for the Natrium reactor were generated using historically-approved protocols and procedures. Although many DBAs without radiological consequence need to be considered, it is impractical to develop a PIRT for each scenario within each event. The DBAs were scrutinized to select three events as representative scenarios, which includes LOOP, RWAP and LOHS as described in the paragraphs below. The phenomena and processes of the selected three representative accidents are considered to encompass the other in-vessel events without radiological release. The three representative events are:

1. LOOP—where two scenarios were examined and PIRTs were performed,
2. RWAP—where two scenarios were examined and PIRTs performed, and
3. LOHS.

The sequences in these representative events used to support the PIRT development, as described in the paragraphs below, may not be identical to those analyzed in the safety analysis.

One of the representative scenarios considered within the LOOP event [[ ]](a)(4) has a sequence of events that includes automatic reactor scram, pump coastdown behavior, reactor transitions to natural circulation, and decay heat removal via the RAC during a long-term cooling period. The LOOP event is initiated with power loss to the scram solenoid valves, the PSPs, and ISPs, causing all control rods to be released and the PSPs and ISPs to coastdown. The NSS isolation valve is closed on loss of power. The RAC system passively removes heat from the reactor.

One of the representative scenarios considered within the RWAP event [[ ]](a)(4) has a different sequence of events that includes positive reactivity insertion, temperature increase in the primary and intermediate loops, and normal heat removal via the intermediate loop. Control rods are assumed to be sequentially withdrawn continuously at the maximum withdrawal rate. It is assumed in this event that the NSS isolation valve is not closed, and the PSPs and ISPs do not trip.

The representative scenario considered within the LOHS event [[ ]](a)(4) is initiated by the loss of power to all ISPs due to a spurious signal. It should be noted that the event sequence is different from the LOOP scenario discussed above. The LOHS event sequence is proposed to represent more appropriate responses as the LOHS event evolves. The ISP pumps are turned off at time zero of the transient. A reactor scram signal is generated on a low ISP flow trip. The PSPs are tripped with the reactor scram to prevent the pump heat from being added to the sodium in the PHT. Natural circulation is initiated in the PHT, and the RAC is operational, removing the decay heat.

A LOOP scenario and an additional RWAP scenario were considered by an internal TerraPower expert panel. The PIRTs for the three scenarios summarized above were performed using an expert panel that was external to TerraPower. The results of these PIRTs are also included in the composite PIRT results.

The five scenarios considered as representative of the above three events were analyzed using the results of representative SAS calculations. For example, the analyses of the scenarios considered by the external expert panel are based on the sequences of events as calculated by the SAS code.

The PIRTs will be updated prior to the FSAR if other events are identified to be representative, or as significant design changes occur.

The protocols and procedures used to develop PIRTs for the above events are described in detail in [5] where the necessary PIRTs were developed as a primary ingredient to Element 1 of RG 1.203. [3]

The PIRTs generated are applicable to Anticipated Operational Occurrences (AOOs), DBEs, DBAs, and BDBEs without fuel failure.

The results of these PIRT studies are captured in high-level summary in the identification of the:

- Phenomena and processes importance rankings
- Knowledge level rankings for the phenomena or processes

The evaluation criteria used to obtain the importance rankings of the phenomena and processes are tied to the FOMs specified in Section 2.2. The FOMs are expected to be re-evaluated as assessment data is collected and evaluated based on the important phenomena and processes identified in the PIRT to ensure that Natrium design acceptance criteria are reflected.

Importance rankings of phenomena/processes identified using their relationships to the FOMs identified above were quantified using the three-level scale shown in Table 2-2. [6] The importance ranking quantifies the level of modeling fidelity required to predict the FOM values as reasonable, as defined in RG 1.203. The importance ranking, therefore, may be regarded as the relative sensitivity of the FOM with respect to the expected variability of the parameters associated with the phenomenon being considered.

Table 2-2. Phenomena/Processes Importance Rankings

Ranking	Description
High (H)	The sensitivity (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.

Three characteristic time periods are considered in evaluating the importance of phenomena/processes. The three time periods are described below.

- Phase 1 (P1, Initiation Phase): from event initiation until the control rods start to drop.
- Phase 2 (P2, Transition Phase): from the time the control rods start to drop, through pump coastdown, until stable natural circulation is attained.
- Phase 3 (P3, Post-scram Cooling Phase): from the time of reaching stable natural circulation flow until decay heat removal by RAC, ambient losses, and/or other systems exceeds generated decay heat, and the long-term cooling stability is sufficiently maintained.

The PIRT panel members identified the phenomena/processes ranking using a defined vote process. [6]

Rankings of the knowledge level of phenomena/processes are made using a three-level scale as 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 relatively low and well characterized.
Medium (M)	The phenomenon is partially known. Data are available but the uncertainties are relatively large.
Low (L)	There is little knowledge regarding the phenomenon. There are high modeling uncertainties.

A knowledge level of high (H) implies additional research on this phenomenon is not necessary even if the importance level is high. Conversely, a knowledge level of low (L) implies that this phenomenon is a priority for additional research, particularly if the importance level is high. A knowledge level of medium (M) implies that research is suggested if the phenomenon is of high importance.

Table 2-4 describes the phenomena and processes identified in the postulated events of the Natrium reactor. The importance and knowledge level rankings of these phenomena are identified in Table 2-5.

Table 2-4. PIRT Phenomena and Processes

[[

System	Component	Phenomena ID	Phenomenon/ Process	Phenomena description

]](a)(4)

[[

System	Component	Phenomena ID	Phenomenon/ Process	Phenomena description

]](a)(4)

[[

System	Component	Phenomena ID	Phenomenon/ Process	Phenomena description

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System	Component	Phenomena ID	Phenomenon/ Process	Phenomena description

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System	Component	Phenomena ID	Phenomenon/ Process	Phenomena description

]](a)(4)

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[[

System	Component	Phenomena ID	Phenomenon/ Process	Phenomena description

]][(a)(4)]

[[

System	Component	Phenomena ID	Phenomenon/ Process	Phenomena description

]](a)(4)

[[

System	Component	Phenomena ID	Phenomenon/ Process	Phenomena description

]](a)(4)

[[

System	Component	Phenomena ID	Phenomenon/ Process	Phenomena description

]](a)(4)

Five preliminary PIRTs were developed for the postulated events, LOOP (1 scenario by a TerraPower PIRT panel and 1 scenario by an external PIRT panel), RWAP (1 scenario by a TerraPower PIRT panel and 1 scenario by an external PIRT panel), and LOHS. Details on the rationale and important discussions relevant to rankings for both the importance and knowledge for individual phenomena and processes are given in [6]. A combined PIRT was generated based on the results of the individual PIRTs for five scenarios while adopting the most conservative ranking among the 5 rankings among individual PIRTs. For the phenomena importance ranking, the higher ranking is more conservative (High > Medium > Low). For the level of knowledge ranking, the lower ranking is considered as more conservative (Low > Medium > High). The combined PIRT (shown in Table 2-5) gives the phenomena/processes rankings together with the knowledge rankings since these items are the basis for determining key physics that must be captured by the EMs and also measured in the experiments designed to generate data for code assessment of the EMs. The PIRT phenomena/processes that are ranked with high importance in phases 1, 2, or 3 are in bold as an indication of their relevance to EM development and assessment.

Table 2-5. Combined PIRT for LOOP, RWAP, and LOHS Licensing Basis Events without Fuel Failure

[[

System	Component	Phenomenon ID	Phenomenon/Process	Importance Ranking			State-of-Knowledge (SOK) Ranking		
				P 1	P2	P3	P 1	P2	P3

]](a)(4)

[[

System	Component	Phenomenon ID	Phenomenon/Process	Importance Ranking			State-of-Knowledge (SOK) Ranking		
				P 1	P2	P3	P 1	P2	P3

]](a)(4)

[[

System	Component	Phenomenon ID	Phenomenon/Process	Importance Ranking			State-of-Knowledge (SOK) Ranking		
				P 1	P2	P3	P 1	P2	P3

]](a)(4)

2.5 Preliminary Evaluation of Highly-Ranked Phenomena

The highly-ranked phenomena identified in the combined PIRT and shown in Table 2-5 may be divided into two groups dependent on whether additional experimental data are required or not. The highly-ranked phenomena for which adequate data already exist are:

- [[

]][(a)(4)]

Because the effect of the above eight highly-ranked phenomena can be quantified using sensitivity studies that envelope the range of interest via input to the EM, these highly-ranked phenomena are input boundary conditions to Element 3 regarding the EM development.

The remaining highly-ranked phenomena are input for consideration of available data from vintage experimental data sets and the design of the TerraPower IET and SETs that are tasks within Element 2.

3 ASSESSMENT BASE DEVELOPMENT: EMDAP ELEMENT 2

The task objectives for Element 2 are focused on obtaining the experimental data necessary "…to provide the basis for development and assessment…" of the EM as described in RG 1.203. The input to Element 2 is that output from Step 4 of Element 1, i.e., the results of the PIRT analysis for scenarios of interest for the Natrium design—as summarized in Chapter 2 of this report.

The output of Element 2 is input to:

- Step 12 of Element 3 to develop and incorporate closure models in the EM and
- Steps 13 through 19 of Element 4 to assess the EM adequacy.

The Element 2 objectives to provide input to Element 3, Step 12 and Element 4 are similar but different as described in Sections 3.1 and 3.2.

3.1 Developmental Assessment: Input to Element 3, Step 12

Step 12 of Element 3 focuses on closure model development and therefore is aimed at ensuring the EM closure models match the physics of the original closure models given in the literature. In addition, the closure models must be demonstrated to properly interact and complement the conservation equations that form the fundamental framework of the EM. Therefore, the input from Element 2 to Element 3, Step 12 is the Development Assessment matrix for the EM. The EM Development Assessment matrix is collection of existing data and calculational assessment problems. The existing developmental assessment matrix is not specific to the Natrium reactor as the matrix addresses a wide operational envelope composed of IETs, SETs, and fundamental physics experiments with a wide range of scales and types; the matrix range is wider than required and defined by the operational and accident envelopes of the Natrium design.

In the process of enhancing the SAS code to include closure relationships required to model the reactor, i.e., the process of creating the EM, the existing Development Assessment matrix will have to be expanded to include data or calculational assessment problems to address the validity of any revisions to the original closure models of the SAS code to create the EM specified to accommodate the required calculational design specific phenomena and processes. [[

]](a)(4)

3.2 Code Adequacy Assessment Matrix: Input to Element 4

The input to Element 4 is the code adequacy assessment matrix and is composed of data sets that are:

- Satisfactory and available in the literature. Satisfactory data sets have a pedigree that describes the experimental facility and test section such that a scalability relationship may be determined (see discussion on data scalability Categories 1, 2, and 3 in Section 3.3), have a data range applicable to the Natrium design, and have measurement uncertainties that are quantified and acceptable.

- Obtained from the IET and SET experimental facilities that are scaled to the Natrium plant. Data from the scaled facilities have the same attributes as described in the previous bullet plus a geometrical scalability with an acceptably low distortion level and compliance with NQA-1 standards.

A flow chart of this process is shown in .

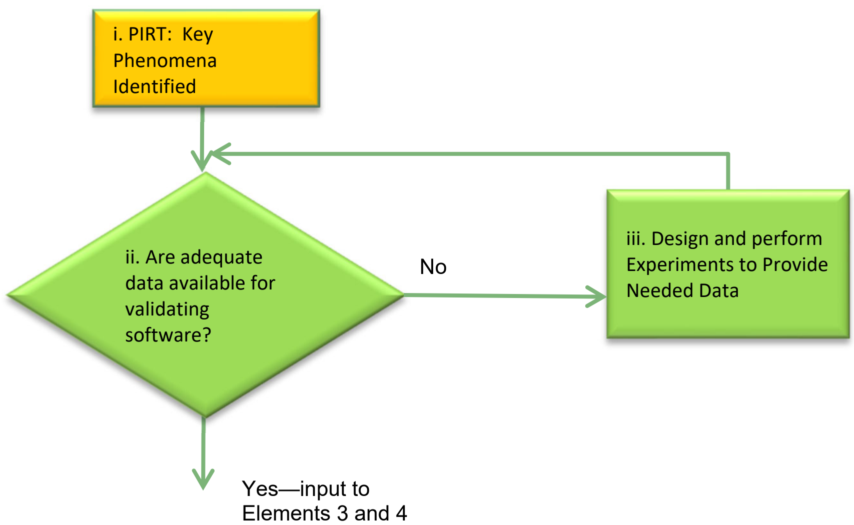

Figure 3-1. Distilled Element 2 flow path.

Sections 3.3 through 3.7 summarize the application of EMDAP Steps 5 through 9 to the EM.

3.3 Assessment Base Objectives: EMDAP Step 5

To determine whether available data are adequate to perform the EMDAP protocols on the Natrium EM, the scalability of the data is considered within 3 categories:

1. Geometry and phenomena: The physical geometry of the experimental facility used to generate data relevant to the Natrium design, including unique features, is assessed considering both the design of the system components and the comparison to experimental facility similarity criteria. All the highly-ranked phenomena defined by the relevant PIRTs must be provided with an acceptable distortion level as defined in the scaling analysis, protocol, and metrics. The working fluid may not be the same as the Natrium working fluid, but if an experiment has a different working fluid than the Natrium plant, then the scaling relationships must accommodate the differences in the working fluid between design and the experiment.
2. Properties: The physical properties (e.g., the thermodynamic state and a similar working fluid), and
3. Phenomena character, event timing, and order: The presence of key phenomena that are projected to be present in the Natrium plant together with similar event timings, ranges, and the order of event

progression. Such data may be found in counterpart tests performed in facilities that have many of the characteristics of the Natrium plant and have different scales from one another including the scaled IET and SET experimental facilities.

It is noted that only an IET and SETs that have been specifically scaled to match geometrically with an acceptably small distortion are able to meet the requirements of Category 1. However, because other IETs, which may be built to achieve other objectives such as proof-of-principle concepts but not to specifically match the Natrium design geometrically with acceptably small distortions, may have scalability from the perspective of Categories 2 and/or 3. For example, if an IET has phenomena and processes that are in some ways similar to the phenomena and processes in the Natrium design, then even though such a facility does not meet the requirements of Category 1 scalability, the data may be used for code assessment because agreement of the EM calculations with the experimental data demonstrates the capability of the EM to calculate the type of phenomena and processes that are projected to be present. A number of counterpart tests that were performed in a variety of scales for similar scenarios and transient conditions satisfy the Category 3 scalability requirements. Examples are data from the EBR-II, FFTF, and Phenix facilities. It is noted that there is historical precedent for including experimental data not only from Category 1 experimental facilities but also Categories 2 and 3 in the EM code assessment matrix.

The objective of the Step 5 tasks is to identify sufficient experimental data to form a complete assessment base for assessing the adequacy of the EM. A complete assessment base has the following characteristics:

- Experimental data from at least one Category 1 IET and the supporting Category 1 SETs deemed necessary are available to support assessment of all the highly-ranked phenomena identified in Element 1.
- Experimental data from other, often legacy, IETs and SETs that may not have an acceptable distortion level to achieve Category 1 scalability requirements, but have many of the geometrical, behavioral characteristics, phenomena, and processes sufficient to qualify as Categories 2 and 3 scalable facilities—provide a medium for establishing credibility for the EM at a variety of scaling factors and conditions that are somewhat different from the typical operational and accident envelope. Using data from such facilities with scaling factors that may differ from that of the Category 1 IET and SETs adds confirmatory evidence of the capability of the EM to perform the required calculations for scenario classes under consideration.

Therefore, the ingredients of the assessment base are obtained from two sources:

- Data from IET facilities and SET facilities scaled to the Natrium plant with acceptable distortion levels and designed specifically to generate data for the highly-ranked phenomena identified in the PIRT.
- Vintage data that may be shown to be similar to the Natrium design.

Figure 3-2 shows the 16 highly-ranked phenomena/processes identified in the PIRT discussed in Section 2.4 relative to the region of interest.

(a)(4)

Figure 3-2. Highly-ranked Phenomena—Relative to the Natrium Design

3.4 Scaling Analysis and Similarity Criteria: EMDAP Step 6

3.4.1 Hierarchical Two-Tiered (H2TS) Scaling

The H2TS scaling methodology serves as the basis to: (a) specify and design the IET and SET experimental facilities with acceptable distortion levels for the specified highly-ranked phenomena and (b) determine the distortion levels, if necessary, for data recorded in legacy experimental facilities. Presently both activities are ongoing and not ready for release. The following discussion summarizes the H2TS methodology to give an example of how it is being used for items (a) and (b). A complete summary of the results of the ongoing scaling analyses will be provided in a later revision to this report.

The challenges associated with hierarchically organized complex thermal hydraulic systems associated with safety issues for nuclear power plants were recognized during the development of the scaling methodology for severe accident analyses in the early 1990s. The hierarchical scaling approach is introduced to combine the system response viewpoint (holistic) and process viewpoint (reductionist) by first describing the hierarchical structure associated with the unique time scales related to the mass/volume ratios, temporal, spatial, and energetics. Two tiers of the methodology are (1) top-down approach to focus on the system response as an aggregate of various processes that take place within a hierarchical level and (2) bottom-up approach to focus on a particular process (prioritized based on their contribution to the system level response). Therefore, a two-tiered scaling approach as part of methodology development guidance addresses the top-down/system-response by efficiency and bottom-up/process-description by sufficiency. Four key elements of the H2TS methodology are described as follows:

(a) System Decomposition, by providing the hierarchical structure of the complex system down to process level description as consistent with the PIRT items.

(b) Identification of Scales (energetic, temporal, spatial scales within each level in the hierarchy)

(c) Top-down/System Scaling Analysis by providing appropriate form of the averaged balance equations for given representative region (or hierarchical level) and deriving the time-ratio groups to determine the scaling hierarchy down to the process-level description.

(d) Bottom-up/Process Scaling by focusing on the processes that have large contributions to the FOM or surrogate FOM such that pedigree, fidelity, and scalability of the models/correlations for the processes are addressed.

The hierarchical decomposition of a given complex system is done first based on the structural/functional description of the system/subsystem/module/components down to a particular volume for which the top-down analysis is to be performed and based on state/process description of the selected volume down to processes contributing to the rate of change in different field variables described by balance equations, i.e., conservation of mass, momentum, and energy. Both decompositions are illustrated in Figure 3-3.

The top-down description of a given hierarchical level to quantify the processes contributing to the rate of change in a given FOM is frequently done through control volume analysis due to its value and flexibility in engineering analysis. Therefore, the rate of change in a given field variable can be determined from the balance equations describing the conservation of mass, momentum, energy, and charge. The control volume analysis is done through averaging of the balance equations over a control volume bounded by a surface across which several transfer/flow paths can be identified by which the communication with other hierarchical levels is established. The averaged general balance can be obtained by deriving the volume-average balance from the local balance such that the rate of change in a specific field variable can be written as follows:

$$\mathcal{M}\frac{d\varphi}{dt} = \sum_m A_m J_m + \sum_j \dot{m}_j \Delta\varphi_j + \sum_n \mathcal{M}_n\, \varphi_n \tag{1}$$

where the volume-averaging symbols are omitted for clarity.

Figure 3-3. Hierarchical Decomposition

In Equation (1) the first term on the right-hand-side of the balance represents the transfer processes that do not involve mass crossing the transfer area, e.g., viscous shear, the second term represents the advection/convection of the conserved property across the flow path or junction, and the last term represents the distributed source/sink mechanisms, e.g., body force (gravity). The control volume balance is written for single-phase material; however, the similar balance can be written for an individual phase/field in a mixture of materials present within the volume. In H2TS methodology, the hierarchy within the volume down to process level is characterized in terms of time-ratio groups which are derived based on the dimensionless form of the balance such that the dimensional analysis is performed on the balance by selecting appropriate reference quantities appearing in the balance. For the field variable, the dimensionless form can be introduced by considering the initial and final values of the variable during the time-interval of interest such that

$$\phi^+ \equiv \frac{\phi - \phi_\infty}{\phi_0 - \phi_\infty} = \frac{\phi - \phi_\infty}{\Delta\phi_\infty}$$

All other quantities appearing in the control volume balance equation are normalized via corresponding reference quantities denoted by subscript (o) such that the dimensionless balance becomes

$$\mathcal{M}^+ \frac{d\phi^+}{dt^+} = \sum_m \Pi_m \cdot \mathrm{J}_m^+ + \sum_j \Pi_j \cdot \dot{m}_j^+ \, \Delta\phi_j^+ + \sum_n \Pi_n \cdot \mathcal{M}_n^+ \Phi_n^+ \tag{2}$$

where Π appears as coefficients for each process contributing to the rate of change in the field variable, ϕ. If the reference quantities are chosen such that $f^+ \equiv f/f_o \approx 1$, the dimensionless rate of change in the field variable can be written as the summation of these Π groups or time-ratio groups such that

$$\frac{d\phi^+}{dt^+} \approx \sum_{i=m,j,n} \Pi_i = \Pi_1 + \Pi_2 + \Pi_3 + \cdots \tag{3}$$

The time-ratio groups can be written for each process as

$$\Pi_i \equiv \omega_i \cdot \tau_o = \frac{\tau_o}{\tau_i} = \frac{\text{System (Control Volume) Time Constant}}{\text{Process } i \text{ Time Constant}} \tag{4}$$

In other words, a time-ratio group compares the individual process time constant to that system to generate the hierarchical structures among the various processes. Furthermore, the processes can be ranked quantitatively according to their importance in the aggregated system response. Therefore, the most dominant process would be the one with

$$\Pi^o \equiv \max_i |\Pi_i|$$

and all the other processes are ranked according to the absolute magnitude of their corresponding time-ratio groups.

3.4.2 Top-down Description of PHT Loop Flow Dynamics

The hierarchical decomposition of the PHT system is performed as shown in Figure 3-4 for the scaling purposes. This decomposition is consistent with the system decomposition discussed in Section 2.3. The first level in the hierarchy considers all the components within the PHT except warm pool and cover gas such that the closed single-phase flow loop is considered. The PHT System is designed to safely remove the heat generated in the core during normal steady-state operation, AOOs, and other off-normal events. The PHT system consists of coolant flow through the reactor core with multiple parallel channels, two identical IHXs, and two identical mechanical pumps. Liquid sodium is discharged from the pumps into the high-pressure plenum, then into multiple core channels composed of fuel, reflector, shield, and control assemblies surrounded by interstitial region. The orifice design at the channel inlets provides flow distribution such that the temperature distribution at the channel exits is fairly uniform.

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Figure 3-4. The Hierarchical Decomposition of the PHT System

The hot liquid sodium from these core channels is mixed in the lower plenum of the hot pool region where the majority of the sodium flows into the region below the bottom of the Upper Internal Structure (UIS) where the control drive mechanisms and fuel handling machine are located. The top hot pool and UIS communicate both thermally and hydraulically. The hot sodium below the sodium/argon interface is fed through the perforated walls into the shell side of the IHX where the heat is transferred into the secondary sodium flowing through the tubes via a counter-current flow configuration between the primary and secondary sides of the heat exchanger. The heat is transferred into the warm sodium pool surrounding the IHX and from the hot pool to the warm pool through the inner vessel liner. The warm pool (listed as “Intermediate Pool” in Figure 3-4) is hydraulically isolated from the hot and cold pools; however, there is a small amount of leakage through the thermal baffles at the top and bottom of the warm pool. The cold sodium exiting the IHX shell-side flows into the cold pool, through the suction pipe into the PSP, and is pumped into the high-pressure plenum through the discharge pipe, completing the flow loop.

The top-down description of the PHT loop dynamics is given based on the closed flow loop schematically shown in Figure 3-5. The flow loop consists of different sections with unique geometry such as flow length (ℓ_i), orientation ($sin\,\theta_i$), flow area ($\mathcal{A}_i$), hydraulic diameter (D_i), and irreversible loss coefficient (K_i). The section orientation angle (θ_i) is defined such that vertically upward sections ($sin\,\theta_i \equiv 1$), vertically downward oriented sections ($sin\,\theta_i \equiv -1$), and horizontal sections ($sin\,\theta_i \equiv 0$)

can be defined. Since the flow loop is a closed loop, the following loop integral should resolve to zero along the flow path, i.e.,

$$\oint_{\text{loop}} sin(\theta)d\ell = \sum_i \ell_i \sin\theta_i = 0$$

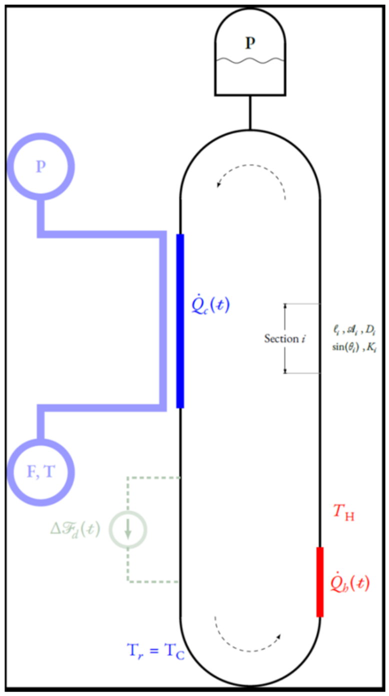

Figure 3-5. Schematic View of a Closed Forced/Natural Circulation Flow Loop

One-dimensional (area-averaged) mass, momentum, and thermal energy equations are assumed applicable in characterizing the single-phase flow around the flow loop depicted in Figure 3-5. This is a valid assumption especially for flow geometries associated with large length-to-diameter ratios, i.e., small aspect ratio. Furthermore, the covariance terms for the velocity-velocity and velocity-temperature are neglected assuming the flow is turbulent with flat velocity/temperature distributions. The one-dimensional characterization of the large pool sections needs to be reevaluated via CFD calculations to represent multi-dimensional effects in the flow/temperature distribution. The wall heat transfer is coupled to the solution of the heat conduction equation with appropriate boundary and initial conditions in the heated and cooled sections. The one-dimensional mass, momentum, and thermal energy equations are summarized as follows: [[

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3.4.3 Establishing Similarity Criteria based on Closed Flow Loop

The governing equations describing the flow loop dynamics coupled with energy balances in each flow segments are described in the previous section. In this section, the governing dimensionless groups are derived by normalizing the balance equations by selecting appropriate reference values for each quantity appearing in the equations. [[

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Table 3-1. Summary of Dimensionless Groups and Their Definitions

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Table 3-2. Similarity Criteria for a Closed Forced/Natural Circulation Flow Loop

Similarity	Criteria
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3.5 Existing Data and SET/IET Needed to Complete Data Base: EM Code Assessment Matrix—EMDAP Step 7

Step 7 consists of three tasks:

1. Construct and perform experiments in the IETs and SETs experimental facilities to create the required database,
2. Identify existing data, and
3. Construct the EM Assessment Matrix.

The following subsections summarize:

- The expectations for obtaining data in IET and SETs scaled to the Natrium design, and the planning that is in progress for them to be designed, built, and operated. The facilities scaled specifically to provide assessment data on the highly-ranked phenomena will provide the backbone of the assessment matrix.

- The vintage reactor facilities and experimental facilities that are candidates for providing assessment data for evaluating the EM adequacy.

Although the scaled IET and SET experimental facilities will provide key assessment data to evaluate the EM adequacy, the data obtained from vintage facilities are essential ingredients to the EM adequacy assessment matrix. The scaled IET and SET facilities are discussed in Section 3.5.1. The vintage IET and SET experimental facilities, from which data sets are presently being considered for inclusion in the EM assessment matrix, are discussed in Sections 3.5.2 through 3.5.12. The pedigrees of the various vintage experimental data sets are discussed in Section 3.5.13. The preliminary EM assessment matrix is given in Section 3.5.14.

3.5.1 Scaled IET and SET Facilities: Category 1 Data

The outputs of Step 6 in Element 2 are scaling analyses and the resulting similarity criteria for IET and SET facilities scaled to the Natrium design with an acceptable distortion level. Such facilities are designed to have the capability to provide data for most of the highly-ranked phenomena and processes that occur in the DBA scenarios identified in this report. A Thermal Hydraulic Testing Roadmap was developed to plan and execute the test campaign, supporting the Natrium plant design and licensing. Some of those test data are used to assess the EM for in-vessel events without radiological release by filling the gap of the phenomena not covered by the historical tests. Presently a single IET is being considered for construction and operation. [[

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Presently four SET facilities are under consideration to obtain data sets related to eight highly-ranked phenomena: [[

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3.5.1.2 [[ ]](a)(4)

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3.5.1.3 [[ ]](a)(4)

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3.5.1.4 [[ ]](a)(4)

The RAC is a passive air-cooling open system during normal and off-normal plant operations. The RAC operates continuously as it is an open system. It requires no power supply and no operator actions as there are no moving parts. RAC heat removal varies with respect to the surface temperature of the GV of the RES. The RES is the heat source that thermally drives air circulation through the RAC system from and to the surrounding environment atmosphere. Since the RAC is an open system, the surrounding environment is both the ultimate coolant source and heat sink for the RAC system.

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3.5.2 EBR-II Tests: SHRT-17, SHRT-45R, and BOP

EBR-II began operating in 1964 and ran until the reactor was shut down in 1994. [13] [14] [15] [16] EBR-II was designed, built, and operated by Argonne National Laboratory. The EBR-II Shutdown Heat Removal Test (SHRT) program was carried out between 1984 and 1986. The objectives of this program were to support U.S. Liquid-Metal Reactor (LMR) plant design, provide test data supporting the validation of computer codes for the design, licensing, and operation of the LMRs, and demonstrate passive reactor shutdown and decay heat removal in response to various transient initiators for both protected and unprotected transient conditions. Among the SHRT tests was a variety of loss of primary and/or intermediate flow tests, loss of heat sink tests, tests to examine the response of the system to changing conditions in the balance of the plant, and tests to characterize reactivity feedbacks.

SHRT-17 was performed during Run 129C and was a protected loss of flow test where a loss of electrical power to all sodium coolant pumps was simulated to demonstrate the effectiveness of natural circulation cooling characteristics. SHRT-45R was performed during Run 138B and was an unprotected loss of flow test where the control rod scram function of the plant protection system was disabled to demonstrate the effectiveness of EBR-II's passive reactivity feedbacks. The Balance of Plant (BOP) series of tests were conducted at EBR-II as part of the SHRT Program. Where the SHRT tests typically examined intentional variations in primary system flow conditions, the BOP tests examined the impact of intermediate system heat removal or core power oscillations on primary system behavior. Table 3-3 shows the evaluation results of the EBR-II tests.

Table 3-3. Pedigree of EBR-II Tests Data

Test	Conducted Under Documented QA Program	Testing Procedure Available	Measured Data Publicly Available	Known Issues With Data	Measurement Uncertainty Quantified
SHRT-17	Unknown, but reference to testing procedures in Reference [14]	Yes, in Reference [15]	Yes, plotted data in Reference [15]	[[	No
SHRT-45R	See above	Yes, in Reference [15]	Yes, plotted in Reference [15]		No
BOP-301	See above	Mostly, defined in Reference [16]	Yes, plotted in Reference [16]		No
BOP-302R	See above	Mostly, defined in Reference [16]	Yes, plotted in Reference [16]	]](a)(4)	No

3.5.3 FFTF Tests: LOFWOS Test #10-12

The Loss of Flow Without Scram (LOFWOS) Test series was conducted at FFTF in 1986 as part of the Passive Safety Testing Program. The LOFWOS test series included thirteen unprotected tests where the plant protection system was intentionally disabled. The goals of the LOFWOS tests were confirming the safety margins of FFTF, providing data for computer code validation, and demonstrating the inherent and passive safety benefits of several of FFTF's design features, such as the limited free core restraint system and the gas expansion modules. [[

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3.5.4 Phenix Tests: Natural Circulation Tests

The Phénix Natural Circulation Test was conducted in 2009 during the End-of-Life Tests Campaign and was designed to represent a protected loss of heat sink with a delayed loss of primary flow with a resumption of secondary system heat rejection. The natural circulation Test was the focus of a large IAEA Coordinated Research Project and is well-documented in reference [17].

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3.5.5 SADHANA Scaled Sodium-Sodium Heat Exchanger Tests

SADHANA Scaled Sodium-Sodium Heat Exchanger Tests. Reference [20] provides a general description of the SADHANA test loop operated by the Indira Gandhi Centre for Atomic Research and mentions scaled sodium-to-sodium heat exchanger tests performed there sometime between 2009 and 2013 to support the design of Indira Gandhi Centre for Atomic Research Prototype Fast Breeder Reactor. References [21] [22] [23] [24] further describe the test facility and present some experimental results.

3.5.6 STELLA-1 Scaled Sodium-Sodium Heat Exchanger Tests

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3.5.7 Toshiba 4S Test Facility Tests

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3.5.8 ANL NSTF Test

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3.5.9 Monju Decay Heat Removal Test

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3.5.10 PCN 37-Pin Bundle Experiments

Multi-subassembly sodium experiments, using the Power Reactor and Nuclear Fuel Development Corporation (PNC) loop PLANDTL-DHX, were performed by PNC in Japan (References [37] and [38]). [[

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3.5.11 WARD 61-Pin Bundle Test

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3.5.12 UIUC Natural Circulation Tests

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3.5.13 Summary of Pedigree Evaluations

The evaluations of the pedigree of historical test data (Non-TerraPower Tests) are summarized in Table 3-4. The first column indicates the test names. The relevancy of the test data to the Natrium reactor, the availability of the data to the public, and the expected data quality are described in the second, third, and fourth columns, respectively. Scales of high (H), medium (M), and low (L) are used for the second and fourth columns. The fifth column provides information on what documentation is available.

Table 3-4. Results of Pedigree Evaluation of Legacy Test Data

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3.5.14 Preliminary Code Assessment Matrix for Natrium EM

Many vintage test cases have been examined, but only a few of them (see Sections 3.5.2 through 3.5.13) have been selected to perform EM adequacy calculations. Also, the plans for designing and constructing scaled IET and SET experimental facilities (see Section 3.5.1) are presently being formulated. Consequently, only a preliminary Natrium EM code assessment matrix is available (see Table 3-5).

[4] Request for data in progress = RDIP

Table 3-5. Preliminary Natrium Code Assessment Matrix

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Note (1): When the legacy test data indicates a gap, the phenomenon will be quantified by performing experiments.

3.6 Evaluation of IET Distortions and SET Scaleup Capability: EMDAP Step 8

The evaluation of IET and SET experimental facilities scaling distortions will be performed based on the magnitudes of the ratios of the similarity criteria identified in Section 3.4.

3.7 Experimental Uncertainties Determination: EMDAP Step 9

Experimental uncertainties associated with vintage data sets were determined by the experimentalists associated with each experimental facility—and as such may not be consistent from one experimental program to another. Also, vintage measurement uncertainties may not be consistent with the NQA-1 standards. Generally, the reported measurement uncertainties for vintage data consist of uncertainties for each measurement type together with approximations of the uncertainties associated with the data acquisition system. No attempts are generally made to separate the uncertainties into aleatory and epistemic components. Therefore, engineering judgement will be applied to the measurement uncertainties of vintage data that are documented to determine the degree of compliance with NQA-1.

The uncertainties of the diagnostics measured and reported for the TerraPower IET and SET experimental facilities scaled to the Natrium design are to be determined in compliance with the NQA-1 standard.

4 EVALUATION MODEL DEVELOPMENT: EMDAP ELEMENT 3

As noted in RG 1.203 [3] (see p. 3):

> "An evaluation model (EM) is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event as illustrated by the following examples:
>
> 1. Procedures for treating the input and output information (particularly the code input arising from the plant geometry and the assumed plant state at transient initiation).
> 2. Specification of those portions of the analysis not included in the computer programs for which alternative approaches are used.
> 3. All other information needed to specify the calculational procedure.
>
> The entirety of an EM ultimately determines whether the results are in compliance with applicable regulations. Therefore, the development, assessment, and review processes must consider the entire EM."

The EM in context of the effort described in this report "is a collection of calculational devices (codes and procedures) developed and organized to meet the requirements established in Element 1" for analyzing in-vessel DBA events without radiological releases. The EM is composed of:

- The SAS4/SASSYS-1 systems code [44] and
- The required input ingredients and post-processing algorithms (e.g., HCF/HPR used to conservatively assess the PCT) that are arranged to model the Natrium plant with the capabilities and reliabilities of the SR SSCs to mitigate and prevent postulated event sequence consequences to within the 10 CFR 50.34 dose limits per NEI-18-04 [2].

Finally, because this report addresses a DBA methodology where a conservative approach is used, an uncertainty methodology is not considered—analogous to the approach defined by the original conservative Appendix K option in 10 CFR 50.46 (see p. A-3 of RG 1.203). Instead, the conservative methodology to be followed is designed to provide sufficient conservatisms without the need for an uncertainty analysis. Therefore, as noted in Appendix A of RG 1.203, "To the extent practicable, predictions of the EM, or portions thereof, shall be compared with applicable experimental information." but without application of an uncertainty methodology.

The above considerations are reflected in the ingredients and requirements defined in the EM development plan (Section 4.1), the EM structure (Section 4.2), and in the development and incorporation of the necessary closure models (Section 4.3).

4.1 EM Development Plan: EMDAP Step 10

The EM development plan includes development standards and procedures that are applied throughout the developmental process per RG 1.203 and in conformance with NUREG-1737: Software Quality Assurance Procedures for U.S. Nuclear Regulatory Commission Thermal-Hydraulic Codes.

[45] In essence, the EM development standards and procedures fall within six areas as identified and summarized below (see RG 1.203 Section 1.3.1, p. 15 and pages B-9 and B-10).

4.1.1 Design Specifications

The specifications are divided into functional requirements, performance requirements, and validation requirements per NUREG-1737. [[

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4.1.1.1 Functional Requirements

Functional requirements consist of the following items:

- The theoretical basis and mathematical models for each phenomenon are shown to be consistent with the subject phenomenon.
- The range of variables over which the model is applied is specified.
- All figures, equations, and references necessary to specify the functional requirements for the design of the software are documented.

4.1.1.2 Performance Requirements

Performance requirements are codified in a software test plan that contains the following considerations:

- The number and type of qualification problems to be performed.
- The rationale for the specification for each qualification problem.
- The specific range of parameters and boundary conditions for which successful execution of the problem set will qualify the code to meet the specific functional requirements.
- Each code input test problem will be described.
- The measure used to determine whether the code results are acceptable will be defined.
- Significant features that will not be tested are identified and the justification for these decisions will be documented.
- The acceptance criteria for each item will be defined.
- The scalability of the qualification problem to the prototype will be identified if applicable.

4.1.1.3 Validation Requirements

All highly-ranked phenomena identified in the PIRT specific to the scenarios of interest must be found to be conservative as measured by reducing the margin of the FOMs.

4.1.1.4 Documentation Requirements

As noted in RG 1.203, p. B-10 “The software design and implementation documentation shall describe the logical structure, information flow, data structures, the subroutine and function calling hierarch, variable definitions, identification of inputs and outputs, and other relevant parameters. It shall include a tree showing the relationship among the modules and a database describing each module, array, variables, and other parameters used among code modules.”

Also, the existing program documentation shall be revised and enhanced to provide a complete description of the program. Code manuals will be produced and upgraded concurrently with the code development process. The set of code manuals, or together with other supplemental documents, will cover the following subjects: Theory, Models & Correlations, User's Manual, Programmer's Manual, and Developmental Assessment Manual.

4.1.1.5 Programming Standards and Practices

The source code listing or update listing shall have the following attributes. Sufficient explanations will be documented in the listing to permit review of these attributes:

- Traceability between the source code and the corresponding design specification to enable analysis of the coding for correctness, consistency, completeness, and accuracy.
- Functionality: Evaluate the coding for correctness, consistency, completeness, accuracy, and testability. Also, evaluation of the design specifications should be enabled for compliance with established standards, practices, and conventions and to enable evaluation of the source code quality.
- Interfaces: Evaluate the coding with hardware, operator, and software interface design documentation for correctness, consistency, and accuracy. At a minimum the ability to analyze data items at each interface will be present.

4.1.1.6 Other Requirements

Transportability requirements, quality assurance procedures, test requirements, and installation requirements.

- Transportability requirements: Thought to be interchangeability between computers and their operating systems.
- Quality assurance procedures: see RG 1.203, Section 2, pp. 20-21.
- Test requirements: All testing activities shall be documented and shall include information on the date of the test, code version tested, test executed, discussion of the test results, and whether the software meets the acceptance test criteria.
- Installation requirements: The program installation package shall consist of program installation procedures, files of the program, selected test cases for use in verifying installation, and expected output from the test cases.

4.1.2 Status of EM Development Plan

The EM development plan has been developed. The EM plan is established by examination of the EMDAP principles and 20 steps, identifying activities necessary to develop the EM, and specifying high-level descriptions of corresponding activities in each EMDAP step. The EM development plan will be updated along with the Natrium reactor's development.

4.2 EM Structure: EMDAP Step 11

The main system analysis computer code of the Natrium EM for the class of scenarios addressed in this report is the SAS code. The structure of the code has already been defined by the SAS development group at the Argonne National Laboratory (ANL). Therefore, this report presents a high-level discussion of the code in Section 4.2.1, and its detailed descriptions can be replaced by referring to the EM manuals. [44]

4.2.1 SAS4A/SASSYS-1 Code Overview

SAS4A/SASSYS 1 is a physics simulation software developed to perform deterministic analysis of anticipated events as well as DBAs in SFRs.

The process to use the SAS code in the form needed to analyze the required DBA analysis scenarios consists of the following:

- Literature research
- Control System Development
- Sensitivity Studies
- Code and Model Benchmarks
- Event Specific Application Methodology
- Sample Events

This approach provides a path that takes advantage of the research and industry experience, yet still allows the development of EM models and nodalization schemes. Therefore, a methodology is devised that meets the requirements of being simple to model yet detailed enough to benchmark against the required experimental data to be used for code assessment.

4.2.1.1 EM Modeling Scope and Limitations

The objective of the EM simulation typically is to quantify accident consequences as measured by the transient behavior of system performance parameters, such as fuel and cladding temperatures, sodium coolant temperatures, pressure, fluid velocities, reactivity, cladding strain, etc. The EM is to perform the safety analysis of the PHT system with heat generation, hydraulic conditions, and thermal conditions for in-vessel DBA scenarios without radiological release. The FOMs that serve as the basis for defining the margin of safety are listed in Table 2-1 and include the fuel centerline temperature, the bulk coolant temperature, and the time-at-temperature criteria..

4.2.1.2 Structure of SAS4A/SASSYS-1

The structure of SAS must contain the following six RG 1.203 ingredients. [3]

- System and components: The SAS structure is designed to enable the analysis of the behavior of all systems and components that describe the physical system of interest.
- Constituents and phases: The models for all the constituents and phases relevant to the required analyses are included in the EM.
- Field equations: The conservation equations that, when solved, calculate the mass, momentum, and energy distribution within the physical system of interest.
- Closure relations: Closure relations are correlations and equations that describe the characteristics of the physical problem that are introduced to obtain a closed solution describing the state of the physical system.
- Numerics: The discretizations of the partial differential equations and closure relationships; the numerical discretizations must be consistent, stable, and convergent.
- Additional features: These address code capability to model boundary conditions and control systems.

High-level discussions of the six constituents of the EM are provided in the following sections. It should be noted that EM is a one-dimensional code (with some zero-dimensional components) and composed of two computer codes, SAS4A and SASSYS-1. SAS4A contains detailed, mechanistic models of transient thermal, hydraulic, neutronic, and mechanical phenomena to describe the response of the reactor core, its coolant, fuel elements, and structural members to accident conditions. SASSYS-1 provides the capability to perform a detailed thermal/hydraulic simulation of the primary and secondary sodium coolant circuits and the balance-of-plant steam/water circuit. Although they are generally portrayed separately, they have always shared a common code architecture, the same data management strategy, and the same core channel representation. The six constituents of the EM are explained in more detail in the code user manual and are expected to be detailed further in a separate report.

a) System and Components

The EM structure was designed to model SFR geometries and thus the systems and components of the Natrium plant. The EM computes coolant pressures, flow rates, and temperatures in the core and heat transport systems. An arbitrary arrangement of components in either a loop-type or a pool-type system can be analyzed. Table 4-1 lists the major components in SAS that are necessary for analyzing the DBA scenarios addressed in this report.

Table 4-1. Geometric Components of SAS4A/SASSYS-1 EM [44]

No.	Component
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The compressible volume (CV) is defined by the CV pressure, volume, mass, and temperature. CVs can accumulate liquid or gas by compressing the cover gas or the liquid, and it is the pressure in the compressible volumes that drives the flows through the liquid and gas segments. CVs are used to model hot pool, cold pool, warm pool, etc. The element, especially the liquid element, is characterized by incompressible single-phase flow, with the exception of the core element. Detailed explanations of the components presented in Table 4-1 including the CV and element are provided in the EM manuals.

In the core models, the basic geometric modeling element is a fuel pin, its cladding, and the associated coolant and structure, with the structure field representing wire wraps, and/or hex cans. In SAS terminology, the term "channel" is used to denote collectively the element consisting of fuel, cladding, coolant, and structure. In a single-pin model, a single average channel is used to represent the average of many pins in the reactor, and multiple channels are used to extend the model to all the pins in the reactor. In a multiple-pin model, each channel represents one or more pins in a subassembly, and multiple-pin subassembly models are joined with single-pin subassembly models to cover the whole reactor core. A single SAS channel may therefore represent either one pin or a large number of pins in many subassemblies. In either case, the elementary unit from a code structure and data management stand-point is an individual channel.

The code structure of SAS is also the result of the programming language employed and the functional requirements of the phenomenological models. The programming language used for SAS4A/SASSYS-1 is ANSI FORTRAN, and the organization of the code follows the FORTRAN convention of the MAIN program with a number of subroutines and functions. For the purpose of this discussion, the subroutines and functions of SAS are grouped according to purpose into one of the modules listed in Table 4-2. These modules are aligned in a one-to-one fashion with the phenomenological models of SAS, each of which is described in the code manuals [44] in detail.

It should be noted that the six modules identified in Table 4-2 are used in DBA in-vessel analyses without radiological release analysis.

Table 4-2. Applicable SAS4A/SASSYS-1 Modules and Phenomenological Models [44]

Module	Purpose/Phenomenological Models
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b) Constituents and Phases

The chemical forms of substance included in the DBA analyses are sodium, air, and argon gas. The sodium for the DBA analysis is in liquid phase. As shown in Table 4-2, the EM has the capabilities to analyze the behavior of all constituents and phases as described in Chapter 2.3. [[
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c) Field Equations

To predict the transport of mass, momentum, and thermal energy of liquid sodium, argon gas, and air present, the EM uses the mass, momentum, and energy conservation equations. Chapter 5 of the SAS4A/SASSYS-1 manuals [44] provides a description of the field equations for the transport of mass, momentum, and thermal energy systems and components except for the core assemblies. Chapter 3 of the manual discusses the field equations used to predict the thermal-hydraulics and thermal conductions of core assemblies separately.

Reactor point kinetics, decay heat, and reactivity feedback models are described in Chapter 4 of the SAS4A/SASSYS-1 manuals and are used to provide an estimate of the reactor power level to be used in the prediction of energy deposition in the fuel. A time-independent reactor power spatial shape is assumed, along with a space-independent (point) reactor kinetics model. The ANS decay heat standard with 23 exponential terms can be used to evaluate decay heat, but SAS uses decay power obtained by curve-fitting the decay power calculated by another code (Burnx). First-order perturbation theory is used to predict reactivity feedback effects associated with material density changes. Fuel temperature (Doppler) effects are calculated assuming a logarithmic dependence on the local absolute temperature ratio, with a linearly dependent variation of the local Doppler coefficient on the coolant void fraction. Besides Doppler and sodium density reactivity feedbacks, axial expansion of the fuel and cladding, core radial expansion, and control rod driveline expansion are also used to calculate reactivity feedback.

d) Closure Relations

Heat transfer correlations (within the fuel pin, between subassemblies, pipe, etc.) and pressure loss are defined by user-supplied coefficients depending on the working fluid and geometry. Thermophysical property correlations of metal fuel, structural metal, sodium, and gas are discussed in the SAS4A/SASYSS-1 manuals in detail.

e) Numerics

Most of the transient heat transfer calculations and flow rate calculations in SAS use semi-implicit time differencing to obtain stable solutions with reasonably long time-steps. Detailed discussions of the numerical solution techniques are presented in the code user manuals Part II, Chapter 3.19 [44].

f) Additional Features

Additional capabilities are available to model control systems. Boundary conditions, steady-state and transient characteristics of special components (pump, valve, etc.), reactor scram, reactivity insertion rate, etc. are modeled using the control system. The control system of SAS consists of four types of signals which are “measured signal”, “demand signal”, “block signal”, and “control signal”. A measured signal makes available to the block diagram the present value of a referenced SAS variable. A demand signal makes available to the block diagram the product of the current value of a time-dependent function defined by the user through a demand table and an initial condition value. A demand table is a set of ordered pair values supplied by the user. A block signal makes available to the block diagram the value at the output of a block. A control signal is used to set the value of a SAS variable equal to the value of a block signal. Again, detailed discussions of the SAS structure are provided in its manuals [44].

g) Software Limitations

The SAS4A/SASSYS-1 is a system analysis computer code developed to model the steady-state and transient system behavior in a pool-type SFR. The usage of the EM for the Natrium design has the following limitations:

- The SAS code is not intended for analyzing the fuel failure and subsequent in-pin or ex-pin fuel relocation, as well as the fission products transport in the sodium pool.
- The application of SAS is limited to single-phase liquid sodium. For sodium boiling, only its impact on reactivity feedback is analyzed, and the impact on fuel/clad heat transfer is not modeled. Sodium freezing is beyond the code capability.
- The software nodalization capability is considered sufficient to model the Natrium plant design, however, nodalization refinement flexibility is limited stemming from some component nodalizations that are hard-wired into the code.
- The software is a 1D system analysis computer code, and thus it is not able to address any 3D effect in the analysis.

4.2.2 EM Structure

The structure of the EM is composed of:

- SAS4A/SASSYS-1
- Data is input to the SAS code that describe fuel performance, neutronics, thermal-hydraulics, design, and safety analysis characteristics are completed.
- A steady-state calculation is performed and a converged solution is obtained.
- The steady-state results are analyzed and a determination is made regarding whether the converged solution is acceptable.
- The desired transient calculation is performed.
- The results of the transient calculation are reviewed and the fidelity of the calculation is assessed.
- The final results are assessed from the perspective of limiting values of the figure-of-merit. Conclusions are formulated.
- The calculation is documented.

This structure is generally illustrated in Figure 4-1.

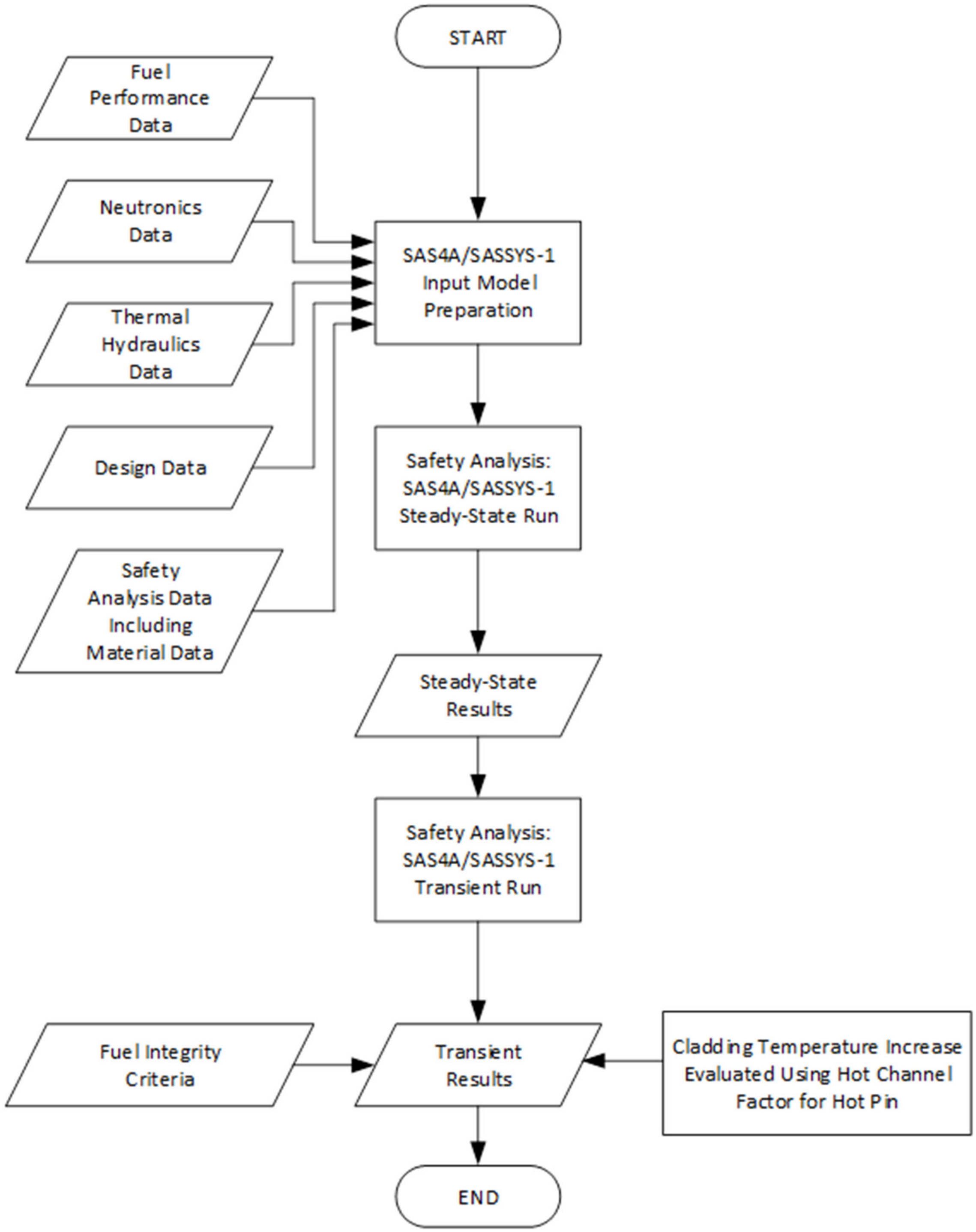

Figure 4-1. EM Structure: Data Inputs, EM Program Flow, and Final Results

4.3 Closure Models and Conservatisms—EMDAP Step 12

Many closure models and conservatisms are used to simulate Natrium responses to postulated DBAs without radiological release. The theory manual of SAS4A/SASSYS-1 [44] discusses the closure models in detail. A summary of the conservatisms that are part of the EM are given in this section.

4.3.1 Closure Models

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RAC is a critical heat sink in many DBAs, especially when the IHX is not fully functioning with the NSS isolation valve closure.

4.3.1.1 Core Convective Heat Transfer

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4.3.1.2 Reynolds-Dependent Pressure Drop

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4.3.1.3 Wire-wrapped Pin-Bundle Pressure Drop

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4.3.2 Conservatisms, Biases, and Hot Channel Factors

Conservatisms are required when performing DBA calculations. Although conservatisms are not closure models, conservative assumptions do affect the outcome of calculations performed using closure relationships and conservatisms are an integral ingredient in the EM. Thus, the conservatisms used for DBA calculations are summarized.

In essence, conservative DBA calculations are performed by revising the input to the nominal best-estimate Natrium model by:

- Inserting conservative biases on the nominal input related to the highly ranked phenomena listed in the PIRT (Table 2-5). Modeling conservatisms are also included directly in the DBA EM such as isolating NSS or tripping the pumps.
- Performing the calculation using the EM to obtain the calculational output.
- Applying the Safety HCF and including the Hot Pin Ratio (HPR) to the output on the channel to obtain conservative 2-sigma cladding temperature.

This process is shown in Figure 4-2 in flow chart form.

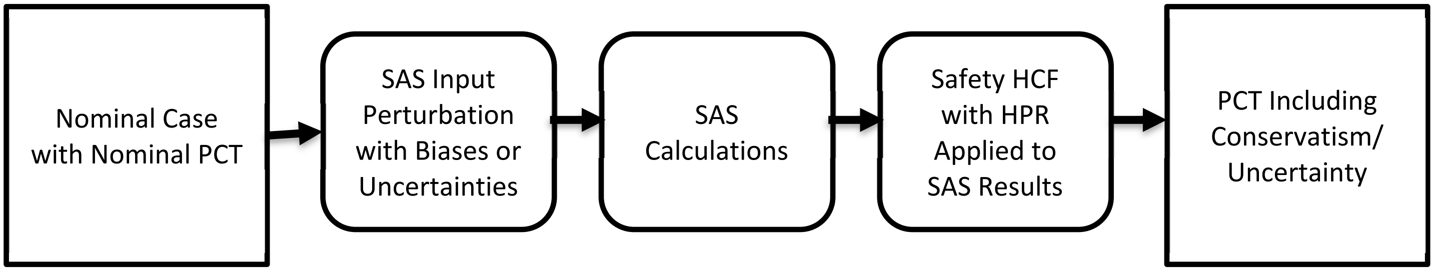

Figure 4-2. Flow chart illustrating methodology for performing conservative calculation of FOM.

EM input biases are applied to the following inputs:

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5 EVALUATION MODEL ADEQUACY ASSESSMENT: EMDAP ELEMENT 4

The information processed in Elements 2 and 3 are used as inputs to Element 4 wherein the adequacy of the EM is assessed. In particular, the specification and implementation of the plans established in Element 3 (Step 10) provide the necessary information to begin the work that comprises Element 4, i.e.:

- A code assessment base was developed (in Element 2) that is consistent with the requirements defined in Element 1. The assessment base consists of already existing experiments or new experiments that serve as a means to determine the adequacy of the EM.
- The EM was developed (in Element 3) to approximate the physical behavior for the postulated events (DBAs for in-vessel events without radiological release) and is consistent with the requirements developed in Element 1. As a part of this task, the proper code options were chosen, the boundary conditions were defined as well as the temporal and spatial relationships among the components.

Element 4 consists of two broad topics:

i. A bottom-up evaluation of the EM (Steps 13 through 15) closure relationships where the closure models and correlations are examined by considering their pedigree, applicability, fidelity to appropriate fundamental or SET data, and scalability and
ii. A top-down evaluation of the code (Steps 16 through 19): the governing equations, numerics, and integrated performance of the EM. Within these stages the EM is evaluated by examining the field equations, numerics, applicability, fidelity to the component and/or IET data, and scalability.

The final step (Step 20) is a consideration of all the outputs of the bottom-up and top-down evaluations performed to determine the EM biases and uncertainties. Each of these steps is described in the subsequent sections.

5.1 Closure Relations (Bottom-up: Pedigree and Applicability): EMDAP Step 13

Step 13 focuses on the pedigree and applicability of the closure relationships used in the EM. A typical closure relationship is the use of a friction factor to approximate the irreversible pressure losses that occur as fluid moves through a pipe—where the magnitude of the friction factor is a function of the roughness of the pipe wall and whether the flow is laminar, in transition from laminar to turbulent, or turbulent. The pedigree and applicability of the friction factor closure relationship consist of the following: (i) documentation: a detailed summary of the experimental work performed to quantify the friction factor including a description of the experimental hardware and instrumentation, i.e., a report or paper available in the literature, (ii) the measurement uncertainty of the instrumentation used to obtain the data, (iii) the range of applicability of the data including the types of fluids for which the data are applicable, e.g., Newtonian fluids, and (iv) the types of hardware for which the data are applicable including how the data may be scaled to different sizes and configurations—in this case, the ratio of the roughness to the pipe diameter, i.e. the relative roughness.

The above approach must be applied and available for all the closure relationships that are used in the EM and the results of the pedigree and applicability studies will be documented in the Models and Correlations document for the EM.

The conclusions and documentation completed in Step 13 are inputs to Step 20 - to determine EM biases.

5.2 Closure Relations (Bottom-up: Model Fidelity and Accuracy): EMDAP Step 14

The model fidelity and accuracy confirmations required in Step 14 are performed by inserting the required input in the SAS code using the guidance given in the Code User's Guide and by performing calculations to demonstrate that the code calculations using the closure relationship match relevant data recorded in experiments that qualify as applicable to the Natrium design for validation purposes—as described in Chapter 4.

To accomplish the objectives inherent to determining the model fidelity and accuracy, the calculations performed to study the closure models should include convergence (discretization) studies that focus on the nodalization (sometimes identified as mesh) that represents the experiments that were built to generate the data underlying the closure model. The discretization studies demonstrate convergence to the calculated results that show agreement with the closure relationship.

Such calculations should be performed for all the closure relationships that are used in the EM when applied to Natrium scenarios. Demonstration of model fidelity and accuracy is shown by reasonable or excellent agreement with the closure relationship predicted results—or if the closure relationship is subjected to a conservative treatment as described in Step 12 then the model should show calculated behavior that demonstrates a conservative outcome, as described in Chapter 5, see discussion regarding Step 10.

This work will be conducted when the experimental data discussed in Section 3.5 become available to TerraPower through performing tests and purchasing tests data and all necessary information. The conclusions and documentation completed in Step 14 are inputs to Step 15 - to assess the scalability of the models.

5.3 Closure Relations (Bottom-up: Assess Scalability of Models): EMDAP Step 15

The scalability of the closure relationships addressed in Step 15 concerns the validity of using closure relationships developed using data from experiments that are a fraction of the size of the Natrium plant. Again, using the example of the friction factor, the use of the relative roughness enables the application of the Moody friction factors [3] over a wide range of pipe sizes. Similar types of scaling relationships should be available and applied for the other closure models, if required, that are used in the SAS4/SASYS-1 code.

Confirmatory calculations or justifications will be conducted when the experimental data discussed in Section 3.5 become available to TerraPower through performing tests and purchasing test data and all necessary information and must be provided for every closure relationship used for calculations.

The conclusions and documentation completed in Step 15 are inputs to Step 20—to determine EM biases.

5.4 Integrated EM - Top-down: Field Equations/Numeric Solutions Capabilities - EMDAP Step 16

The objective of Step 16 is to determine the capability of the field equations to represent processes and phenomena as well as the ability of the numeric solutions to approximate the equation set. The SAS4A/SASSYS-1 code field equations, i.e., the conservation equations of mass, momentum, and energy are discretized using finite difference equations (FDEs). The partial differential equations (PDEs) themselves have been derived, in general, to describe single-phase flow and the EM may be used to analyze the behavior of systems with both water and liquid sodium working fluids.

Based on 10 CFR 50.46 Appendix K, the momentum equation needs to have accommodations to satisfy the need to calculate the following effects: (1) temporal change of momentum, (2) momentum convection, (3) area change momentum flux. It needs to be capable of accommodating the need to determine the energy transfer and distribution within the fuel as well as from the fuel to the working fluid, to the reactor components and structures, and to the environment. The SAS code will be evaluated against the above requirements.

The validity of the PDEs will be demonstrated by performing validation calculations using data from experiments that are scaled as well as experiments that have partial scalability to the Natrium plant.

In general, the scalability of the data to the Natrium design is considered within 3 categories:

1. Geometry & phenomena: The physical geometry of an experimental facility used to generate data relevant to the Natrium design including unique features of the Natrium design assessed considering both the design of the system components and the comparison to experimental facility similarity criteria. Data for all of the highly-ranked phenomena defined by the relevant PIRTs must be provided with an acceptable distortion level.
2. Properties: The physical properties (e.g., the thermodynamic state and a similar working fluid), and
3. Phenomena character, event timing and order: The presence of key phenomena that are projected to be present in the Natrium plant together with relevant event timing and the order of event progression.

Only an IET facility scaled to represent the Natrium design with an acceptable distortion level can satisfy all 3 scalability categories. Other experimental facilities, e.g., EBR-II may satisfy scalability categories 2 and 3. Similarly, SET facilities will consist of vintage facilities that satisfy particular data needs as well as newly designed SET facilities built to satisfy specific data needs.

The relevance of the field equations in the SAS code is shown by the pedigree, key concepts, and processes that are characteristic of the SAS computer code. In essence, the SAS code was developed specifically to analyze the behavior of SFRs. Consequently, the historical development and evolution of the code reflect the creation of specific components designed to represent a pool-type SFR. These characteristics will be distilled from the existing documentation and included in subsequent revisions of this topical report as well as in the manuals being written to satisfy the RG 1.203 requirements for computer code manuals (see Appendix B, p. B-9 and B-10) such as the Theory Manual and the Developmental Assessment manual. The validation cases in the Developmental Assessment manual of interest are those that specifically satisfy Categories 2 and 3 of the scalability requirements described above.

The numeric solution evaluation considers consistency, property conservation, and stability of the SAS code. In essence, consistency is characterized by the extent to which the discretized equations approximate the partial differential equations. An FDE representation of a PDE is considered consistent if it can be shown that the difference between the PDE and the FDE and its difference representation vanishes as the mesh is refined, that is:

$$\lim_{h\to 0}[PDE - FDE] = \lim_{h\to 0}[TE] = 0$$

as the spatial mesh interval h approaches zero for both [PDE – FDE] and the truncation error [44].

The conclusions and documentation completed in Step 16 are inputs to Step 17—to determine the capability of the EM to simulate system components.

5.5 Integrated EM - Top-down: Assess Applicability of EM to Simulate System and Global Capability: EMDAP Steps 17 and 18

Steps 17 and 18 of RG 1.203 are specified to first evaluate the inherent capability of the EM to model the major systems and subsystems of the Natrium design and second to assess the system interactions and global capabilities of the EM. The historic work described in the pedigree documentation—reported upon and considered in Section 5.4—demonstrates the capabilities of the SAS4A/SASSYS-1 code to reasonably model SFRs. The commercial grade dedication (CGD) for the SAS4A/SASSYS-1 is performed following the guidance of the EPRI report 3002002289 [49]. Software technical evaluation [46], acceptance test plan [50], acceptance test report [51], and the final summary and conclusion of CGD is documented in the SAS4A/SASSYS-1 Software Dedication Report [52]. The CGD will be performed if a version of the code is adopted for the application. The conclusions and documentation to be completed in Step 17 are inputs to Step 20—to determine EM biases.

The assessment of the system interactions and global capabilities of the EM focus on the fidelity of calculations performed using the EM. The demonstration of the EM fidelity is accomplished by satisfactory completion of the following tasks:

- Identification of the optimal model representation of Natrium plant components and system.
- Confirmation of a nodalization (mesh) that gives convergent solutions for both the Natrium plant and the models used to perform the validation studies using the experimental data sets that make up the validation matrix.
- Application of the same model options and nodalization in both the Natrium design and experiment validation calculations.
- Assessment and confirmation that all the highly-ranked phenomena identified in the PIRT are calculated in either a reasonable or excellent fashion for a best-estimate calculation, or are suitably conservative.
- Quantification of the biases and deviations of the validation calculations and the subject validation data.
- Evaluation of the ability of the EM to model system interactions, e.g., between the loops inherent to the heat exchangers. This objective is achieved by comparing the calculated interactions between system components that are present in the scaled IET experiments.
- Quantification of the parameter ranges characteristic of the Natrium plant for the scenarios under consideration.

Upon satisfactory completion of the above tasks, the final step of the integrated EM adequacy may proceed. The conclusions and documentation completed in Step 18 are inputs to Step 19—to assess the scalability of the integrated calculations and the distortion level of the experimental data.

5.6 Integrated EM - Top-down: Scalability Assessment of the Integrated EM: EMDAP Step 19

The scalability assessment of the integrated EM is performed in conjunction with the scalability assessment of the closure models (Step 15). The results of the two scalability assessments are integrated and conclusions are formed for consideration in Step 20.

From an integrated EM perspective, the scaling assessment consists of ensuring that the agreement between the experimental data and the EM calculations of the highly-ranked phenomena identified in the Natrium PIRT studies (considered in Element 1 of RG 1.203) is reasonable at a minimum, i.e. sufficiently conservative—together with an assessment of the distortion level of the measured data. Provided the distortion levels are acceptable, following evaluation viz-a-viz the plan requirements (Step 10), the conclusions formulated are one of the primary input ingredients to Step 20.

5.7 Determine EM Biases and Uncertainties: EMDAP Step 20

Because the methodology used for the evaluation of DBAs for in-vessel events without radiological release is conservative, no uncertainty analyses are required. Instead, a conservative approach is being defined and an effort is underway to demonstrate that it is "suitably conservative."

6 NATRIUM SAMPLE ANALYSIS RESULTS

At the time of this writing, the majority of DBA analyses have not been performed in sufficient detail to warrant inclusion in this report. Sample DBA evaluations will be performed and documented prior to submitting a final update of this evaluation methodology.

7 ADEQUACY DECISION

The adequacy decision provides documentation 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 DBA methodology documented in this report for use by future applicants utilizing the Natrium design as an appropriate and adequate means to evaluate in-vessel DBA events without radiological release. This approval is subject to the limitations described below.

8.2 Limitations

This section describes the limitations of the 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. Changes from these design features will be identified and justified in Safety Analysis Reports of Natrium license applications.

2. 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 EM. This verification and validation information should be justified to reasonably bound the operational envelope for the design for any applicant referencing the in-vessel DBA EM methodology.

3. An applicant utilizing the topical report needs to justify the use of the model for the design. This justification must discuss the capability of the model in the context of what is needed to appropriately represent the design and discuss how the model is applicable to the design, including consideration of system interactions occurring in the design, system conditions (which may affect the applicability of models or validation data). If the design requesting use of this model shows significant change from the design described in this report, a revised PIRT (or functionally similar tool) should be made available to facilitate review of the final validation model. Demonstration of the use of suitably conversative methods will be provided, or uncertainties associated with the evaluation model and the validation data should be discussed in accordance with RG 1.203.

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