February 28, 2025

TP-LIC-LET-0398
Docket Number 50-613

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 Events with Radiological Release," NAT-9394 Revision 0

**Reference:** 1. Transmittal of TerraPower, LLC Topical Report, "Design Basis Accident Methodology for Events with Radiological Release," Revision 0, March 22, 2024 (Accession No. ML24082A262)
2. TerraPower, LLC – Audit Plan for Topical Report TP-LIC-RPT-0007, "Design Basis Accident Methodology for Events with Radiological Release," Revision 0, July 15, 2024 (Accession No. ML24197A156)

This letter transmits the TerraPower, LLC (TerraPower) Topical Report, "Design Basis Accident Methodology for Events with Radiological Release", NAT-9394 Revision 0 (enclosed). The report contains an overview and description of the model developed to evaluate Design Basis Accident events with the potential for radiological release for the Natrium® Plant[1]. The revised report is provided to replace the information contained in Reference 1 with supplemental detail to address questions discussed during the NRC audit of the methodology (Reference 2), and includes other editorial revisions.

As described in Reference 1, the purpose of submitting this Topical Report is to provide information to the U.S. Nuclear Regulatory Commission (NRC) to facilitate efficient and timely review of the TerraPower Design Basis Accident Methodology for Events with Radiological Release. TerraPower also requests, as part of this review and associated comment resolution, that the NRC provide a safety evaluation report (SER) on the methodology.

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

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 only be disclosed to Foreign Nationals in accordance with the requirements of 15 CFR 730 and 10 CFR 810, as applicable. Proprietary and ECI materials have been redacted from the report provided in Enclosure 2; redacted information is identified using [[ ]]$^{(a)(4)}$, [[ ]]$^{ECI}$, or [[ ]]$^{(a)(4), ECI}$.

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

If you have any questions regarding this submittal, please contact Ian Gifford at igifford@terrapower.com.

Sincerely,

George Wilson

George Wilson
Senior Vice President, Regulatory Affairs
TerraPower, LLC

Enclosures:
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 Events with Radiological Release," NAT-9394 Revision 0 – Non-Proprietary (Public)
3. TerraPower, LLC Topical Report, "Design Basis Accident Methodology for Events with Radiological Release," NAT-9394 Revision 0 – Proprietary (Non-Public)

cc: Mallecia Sutton, NRC
Josh Borromeo, 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 Senior 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: February 28, 2025

George Wilson

*George Wilson*
Senior Vice President, Regulatory Affairs
TerraPower, LLC

**ENCLOSURE 2**

**TerraPower, LLC Topical Report,**
**"Design Basis Accident Methodology for Events with Radiological Release," NAT-9394 Revision 0**

**Non-Proprietary (Public)**

*Verify Current Revision*

| | | | |
|---|---|---|---|
| **Document Title:**<br>Design Basis Accident Methodology for Events with Radiological Release | | | |
| **Natrium Document No.:**<br>NAT-9394 | **Rev. No.:**<br>0 | **Page:**<br>1 of 65 | **Target Quality Level:**<br>N/A |
| **Alternate Document No.:**<br>NAT-9394-NP | **Alt Rev.:**<br>N/A | **Originating Organization:**<br>TerraPower, LLC (TP) | **Quality Level:**<br>N/A |
| **Natrium MSL ID:**<br>N/A | **Status:**<br>Released | | **Open Items?**<br>0 |
| **Approval** | | | |
| Approval signatures are captured and maintained electronically; see Electronic Approval Records in EDMS. Signatures or Facsimile of Electronic Approval Record attached to document. | | | |

# REVISION HISTORY

| Revision No. | Affected Section(s) | Description of Change(s) |
|---|---|---|
| 0 | All | Initial Release – Supersedes TP-LIC-RPT-0007 Rev. 0. Incorporates changes made to address NRC questions during audit review. Changes from previous information marked via change bars. |
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## ACRONYMS

| | |
|---|---|
| ANL | Argonne National Laboratory |
| AOO | Anticipated Operational Occurrence |
| ATR | Advanced Test Reactor |
| BDBE | Beyond Design Basis Event |
| CDF | Cumulative Damage Fraction |
| CFD | Computational Fluid Dynamics |
| CGD | Commercial Grade Dedication |
| CP | Construction Permit |
| DBA | Design Basis Accident |
| DBE | Design Basis Event |
| DOE | Department of Energy |
| DSAW | Detailed Safety Analysis Workflow |
| DSC | Differential Scanning Calorimetric |
| EBR-II | Experimental Breeder Reactor II |
| EM | Evaluation Model |
| EPZ | Emergency Planning Zone |
| EVHM | Ex-Vessel Handling Machine |
| EVST | Ex-vessel Storage Tank |
| F-C | Frequency-Consequence |
| FCCI | Fuel Cladding Chemical Interaction |
| FEA | Finite Element Analysis |
| FH | Fuel Handling |
| FHA | Fuel Handling Accident |
| FHB | Fuel Handling Building |
| FOM | Figure of Merit |
| HAA | Head Access Area |
| HCF | Hot Channel Factor |
| IAC | Intermediate Air Cooling |
| IET | Integral Effects Test |
| IHT | Intermediate Heat Transport System |
| INL | Idaho National Laboratory |
| ISP | Intermediate Sodium Pump |
| IVTM | In-Vessel Transfer Machine |
| LBE | Licensing Basis Event |
| LHGR | Linear Heat Generation Rate |
| LHR | Linear Heat Rate |
| LWR | Light Water Reactor |
| MFF | Mechanistic Fuel Failure |
| NI | Nuclear Island |
| NRC | Nuclear Regulatory Commission |

| | |
|---|---|
| NSRST | Non-Safety-Related with Special Treatment |
| NST | No Special Treatment |
| OQE | Other Quantified Event |
| ORNL | Oak Ridge National Laboratory |
| PCT | Peak Cladding Temperature |
| PHT | Primary Heat Transport System |
| PIC | Pool Immersion Cell |
| PIRT | Phenomena Identification and Ranking Table |
| PSAR | Preliminary Safety Analysis Report |
| QA | Quality Assurance |
| QAPD | Quality Assurance Program Description |
| RAB | Reactor Auxiliary Building |
| RAC | Reactor Air Cooling System |
| RCC | Reactor Core System |
| RES | Reactor Enclosure System |
| RN | Radionuclide |
| RSF | Required Safety Function |
| RV | Reactor Vessel |
| RVH | Reactor Vessel Head |
| RWG | Gaseous Rad Waste Processing System |
| RXB | Reactor Building |
| [[ | ]](a)(4) |
| SCG | Sodium Cover Gas System |
| SFP | Spent Fuel Pool |
| SFR | Sodium-Cooled Fast Reactor |
| SMP | Software Management Procedure |
| SPS | Sodium Processing System |
| SQA | Software Quality Assurance |
| SR | Safety Related |
| SSCs | Structures, Systems, and Components |
| TATNF | Time at Temperature No Failure |
| TREAT | Transient Reactor Test Facility |
| ULOF | Unprotected Loss of Flow |
| ULOHS | Unprotected Loss of Heat Sink |
| UTOP | Unprotected Transient Over Power |
| V&V | Verification and Validation |

# EXECUTIVE SUMMARY

This topical report provides a high-level road map for, and summary of, the Design Basis Accident Methodology for Events with Radiological Release for the Natrium™ reactor, a TerraPower & GE-Hitachi Technology. It describes the evaluation model (EM) development, the resulting EMs, and identifies EM items which require further development. Certain aspects of the EM adequacy demonstration remain in development and are noted throughout the report. It is acknowledged that this report contains preliminary technical information, and several sections within describe future actions that are planned to be taken by TerraPower. Information generated by these actions will be provided in future licensing submittals. These actions are expected to be complete prior to use of this EM in support of an operating license application.

This report contains six chapters and two appendices.

Chapter 1 discusses the overall objective and scope of the report.

Chapter 2 discusses the regulatory requirements and guidance used in the EM development process, and a high-level description of the Natrium nuclear power plant. Chapter 2 also identifies the safety systems and design basis accidents that pertain to the Design Basis Accident with radiological release EM development.

Chapter 3 lists Assumptions and Open Items.

Chapter 4 discusses the general EM requirements, the independently submitted topical reports that are utilized, and the capability development for analysis of different design basis accident (DBA) with release scenarios.

Chapter 5 discusses the event-specific EMs/methodologies established for analysis of

- In-vessel transients with radiological release (Section 5.1)
- Partial flow blockage (Section 5.2)
- Fuel misload (Section 5.3)
- Fuel handling accidents (Section 5.4)
- Sodium liquid and gas leaks (Section 5.5)

Each section includes the following subsections:

- Purpose and scope
- Assumptions
- EM scope and requirements
- EM descriptions
- EM assessment

Chapter 6 provides some conclusions on the EM development and summarizes the limitations and conservatisms of the EMs.

Appendix A provides details on the use of Time at Temperature No Failure (TATNF) and related analyses.

Appendix B provides a list of legacy experimental data available for EM verification and validation.

# 1 PURPOSE

This topical report addresses the Natrium™ nuclear power plant DBA with radiological release EM development process, the resulting EM, and identifies EM development items which require further development. The methodology development guidance provided in the Natrium Reactor Project General Methodology Development and Assessment Guide, was used in the development of this EM.

The Natrium power plant being developed by TerraPower follows the methodology provided in NEI 18-04, Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development, to identify and evaluate Licensing Basis Events (LBEs) including frequency based Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), Beyond Design Basis Events (BDBEs), and conservative assumption oriented DBAs [1]. Additionally, the identification and classification of safety-related (SR) and non-safety-related with special treatment (NSRST) structures, systems, and components (SSCs) are determined consistent with the methodology presented in NEI 18-04. Figure 1-1 provides a graphical representation showing the AOO, DBE, BDBE, and DBA relationships as well as how they fit within the complete event structure from a frequency perspective.

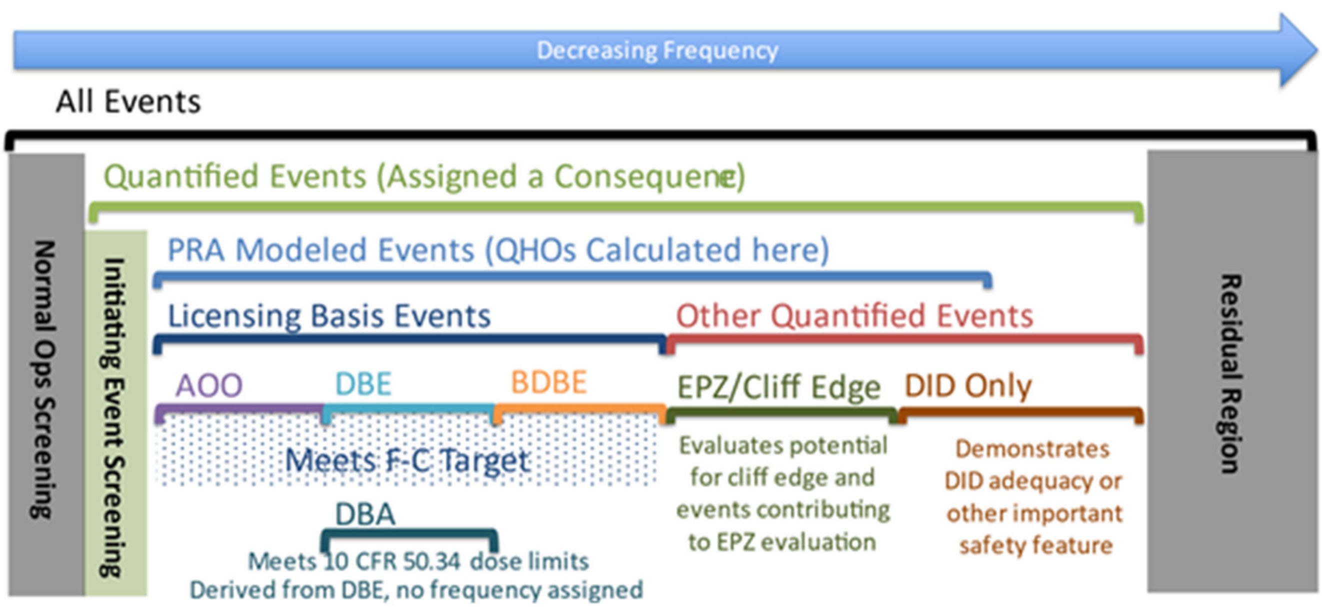


**Figure 1-1. Frequency oriented relationship between AOOs, DBEs, BDBEs, and DBAs.**

The guidance provided in NEI 21-07 - Technology Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: For Applicants Utilizing NEI 18-04 methodology - is followed in the development of the Natrium Preliminary Safety Analysis Report (PSAR) [2]. The PSAR is being developed in accordance with the two-part licensing approach established in 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, which involves first obtaining a Construction Permit (CP) followed by an Operating License. The PSAR is submitted to the Nuclear Regulatory Commission (NRC) as part of the CP application process. It is important to note that the PSAR will necessarily contain preliminary design information which must be updated as the process reaches conclusion, and an Operating License is requested.

NEI 21-07 states the following with respect to DBA analytical method discussion in the PSAR,

> *The applicant should describe the overall analytical methodology and identify and describe the significant computer codes used to model the plant response. The applicant should address the applicability of the analytical methodology to the characteristics of the plant, including a discussion of the underlying experimental or analytical basis. Typically, this is done through NRC-reviewed and approved topical reports that are incorporated by reference in the SAR or through technical reports that are summarized in the SAR and available for regulatory audits.*

To support development of the PSAR, this report provides discussion of the evaluation model development used to evaluate the Natrium plant response where the release of radioactive material is a possible consequence of a DBA. Furthermore, consistent with NEI 21-07, for these scenarios, a mechanistic source term is used in the calculation of the consequences. This report provides a high-level discussion of the following issues associated with the evaluation model development for Natrium DBAs with radioactive material release:

- DBA event selection
- Important processes and phenomena
- Overall analytical methodology
- Identification and description of significant computer codes used to model the plant responses
- Applicability of the analytical methodology to the characteristics of the plant
- Underlying experimental or analytical basis for model assessment and model pedigree

In the Natrium plant, DBA scenarios can be grouped into two basic physical areas: in-vessel scenarios and ex-vessel scenarios.

In-vessel scenarios include traditional reactor transient scenarios leading to fuel damage and subsequent release of radioactive material. Furthermore, in-vessel scenarios include flow blockage scenarios, fuel handling accident scenarios and loss of active cooling scenarios. The principal difference between reactor transients and the other in-vessel scenarios involves the use of [[ ]](a)(4) to analyze the reactor transient phenomena while the other scenarios do not require this code system.

Ex-vessel scenarios include fuel handling accident scenarios, loss of active cooling scenarios, and radioactive sodium and gas leak scenarios. Apart from [[ ]](a)(4) which is not used for ex-vessel scenarios involving the release of radioactive material, these scenarios require a similar set of computer codes that are used for the analysis of in-vessel scenarios. The codes are noted and discussed throughout this report.

# 2 BACKGROUND

## 2.1 Regulatory Requirements and Guidance for DBAs

DBA postulated accidents are used to set design criteria and limits for the design and sizing of safety-related systems and components. 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 NEI 18-04 is:

> *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.*

## 2.2 Plant Description

The Natrium Reactor is a sodium-cooled fast reactor (SFR) that uses a fuel design and an operating environment that are significantly different from light water reactors 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 Natrium Reactor 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 Power Reactor Innovative Small Module technology and TerraPower's Traveling Wave Reactor technology.

The general plant layout is shown in Figure 2.2-1 and is made up of two basic areas; a Nuclear Island where the reactor and associated support facilities reside and an Energy Island where thermal storage tanks and turbine facilities for generating electricity reside. Safety functions are made integral to the reactor vessel and support equipment is moved to separate structures in the Energy Island, resulting in a simplified reactor building. Decoupling the Nuclear Island from the Energy Island from a nuclear safety perspective is central to simplifying the Natrium design. The Natrium design capitalizes on the proven metal fueled SFR safety characteristics to minimize the number of safety-related SSCs needed to achieve safety goals.

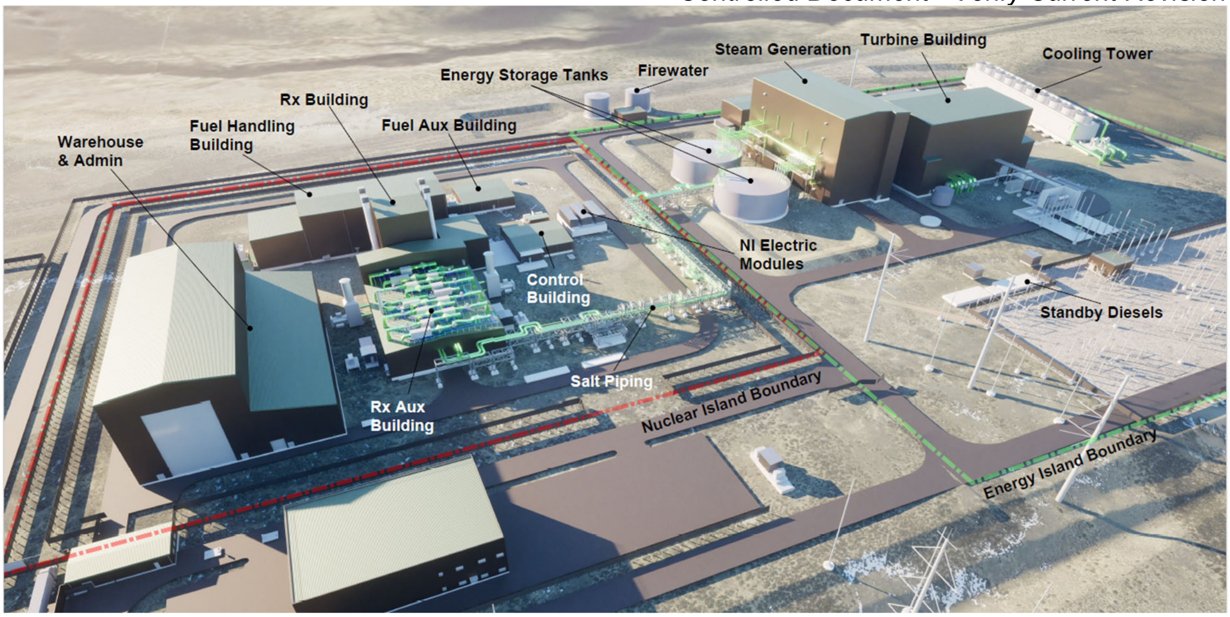


**Figure 2.2-1. Plant Layout.**

The Natrium plant uses a pool-type design with the reactor core and primary coolant pumps located within a large pool of primary sodium coolant and no penetration through the reactor vessel thereby eliminating loss of coolant accidents involving primary pumps and piping. The primary sodium pool operates at near atmospheric pressure. Heat is transferred from the hot primary sodium pool to an intermediate sodium piping loop by means of two intermediate heat exchangers. The intermediate piping loop uses non-radioactive sodium to transport reactor heat from each intermediate heat exchanger to two sodium/salt heat exchangers. These sodium/salt heat exchangers in the Nuclear Island heat salt received from the cold salt tank in the Energy Island. The heated salt is then returned to the Energy Island for storage in the hot salt tank, which serves as thermal energy storage. The salt stored in the hot tank is used to generate steam for use in steam turbine generators eliminating the need for generating steam directly from reactive sodium metal. The Natrium plant can vary its supply of energy to the grid through its energy storage system. The Natrium reactor operates at a thermal power of 840 MW while the plant produces 336 MWe steady-state and 500 MWe peak power. The thermal energy storage system, located in the Energy Island, uses two molten salt tanks, one hot and one cold. Its architecture is like molten salt systems for concentrated solar power. The charging salt loop transports salt from the cold tank to the reactor for heating and routes it to the hot tank. The steam / salt loop transports salt from the hot tank to steam generators to generate superheated steam and returns salt to the cold tank.

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

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

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

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

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

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

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

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

The Nuclear Island is composed of six major buildings: reactor, fuel handling, control, electrical, reactor auxiliary, and fuel auxiliary buildings. The reactor building, see Figure 2.2-2, houses two major components: the reactor and RAC air ducts. The reactor is located below grade to protect it from natural hazards (earthquakes, tornadoes, etc.) and other hazards. There are only two rooms in the reactor building, 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 reactor building below ground to the reactor auxiliary building.

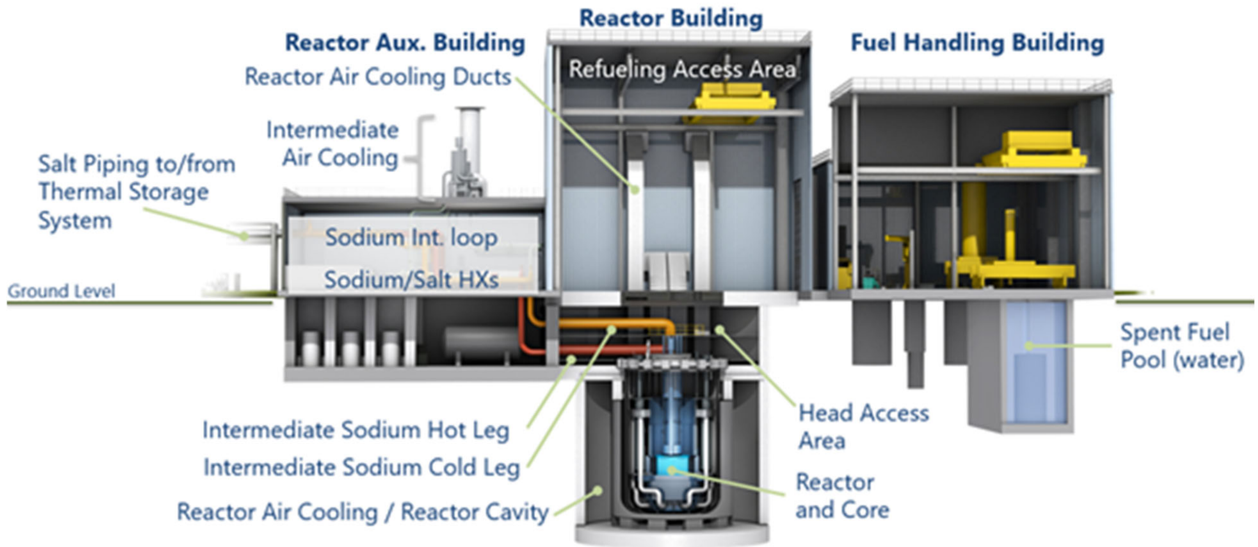


**Figure 2.2-2. Natrium Elevation View.**

The fuel handling building 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 reactor building to the fuel handling building. The buildings are connected by a rail system at ground level to support movement of the fuel handling cask. The fuel handling building 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 Nuclear Island (NI) Control Building uses a structural steel braced frame supported on a concrete grade slab with insulated metal siding and an insulated standing seam metal roofing or membrane roofing system. During normal operations, systems will be monitored and controlled from this building.

The Reactor Enclosure System (RES) contains and supports the reactor core and primary sodium coolant, including all supporting equipment and structures. The RES is divided into five subsystems: Reactor Vessel (RV), reactor internals, Reactor Vessel Head (RVH), Guard Vessel, and Reactor Support Assemblies. All subsystems are in, and are either directly or indirectly supported by, the Reactor Building. The RV, along with the RVH, form most of the reactor coolant and primary cover gas boundaries. Finally, the RV and RVH provide support for the reactor internals as well as the Core Support Structure, which supports the reactor core.

The In-Vessel Transfer Machine (IVTM) moves core assemblies between the core, in-vessel fuel storage racks, and transfer station for removal from the reactor vessel. 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 reactor vessel with the fuel transfer lift operating through the reactor transfer adapter. Fresh core assemblies are transferred into the fuel transfer lift and are then lowered into the pool region by the fuel transfer lift to core level to be transferred into the core using the IVTM. Used core assemblies are transferred out of the core to the in-vessel storage for decay or directly to the fuel transfer lift for assemblies which do not require in-vessel decay. The IVTM and fuel transfer lift are installed at the beginning of a refueling outage, the IVTM installed on the rotating plug assembly, and the fuel transfer lift penetrating the reactor vessel head. They make up part of the functional containment boundary during refueling operations and are removed after refueling is complete.

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

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

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

## 2.3 Safety System Classification

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

### Safety-Related (SR)

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

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

**Non-Safety-Related with Special Treatment (NSRST)**

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

**Non-Safety-Related with No Special Treatment (NST)**

All other SSCs (with no special treatment required).

## 2.4 DBA Event Selection

The DBAs identified for the Natrium design can be broadly categorized as:

- In-vessel core transients with fuel failure includes symmetric and asymmetric Primary Heat Transport System (PHT) and Intermediate Heat Transport System (IHT) initiated events, the loss of hydraulic holddown, RAC long-term transient, etc.
- Local faults (including partial flow blockage and fuel misload)
- Fuel handling events
- Radioactive gas/liquid leakage/release events.

Of the DBAs identified, ten of the DBAs have descriptions indicating they involve a potential release of radioactive material and are listed in Table 2-1. It should be noted that the list of DBAs in Table 2-1 represent the currently identified events and are provided to help illustrate the methodology in the report only, not to define the set of events applicable to all Natrium plants. Applications incorporating this report by reference will utilize the methodology outlined in the report for the relevant events defined in the application. To determine which DBAs have the potential for release of radioactive material, the following process is used. [[

]](a)(4) Once the DBAs are established, they are distributed to the functional groups for evaluation.

In-vessel DBAs are first analyzed with the in-vessel DBA without release EM [3] to obtain cladding temperature results. These results are then compared to the TATNF screening criteria. DBAs which do not violate the TATNF screening criteria are then excluded from further consideration of potential release. Those that violate the TATNF screening criteria are taken for further evaluation with the Detailed Safety Analysis Workflow (DSAW) process as described in Section 5.1.5.

DBAs which are not in-vessel are evaluated using the appropriate methodology described below, an appropriate event-specific method, or evaluated with the source term EM using conservative assumptions. Thus, essentially, any DBA which is not precluded from having releases based on the DBA without release EM and TATNF screening criteria are assumed to have the potential for radionuclide release.

Note that the following events in Table 2-1 do not have a corresponding EM within this topical report:

- [[ ]](a)(4): There is not a specific EM within this topical report for this event to determine the extent of fuel failure due to an excessive sodium chemical reaction. The Radiological Source Term Methodology [4] simply takes a conservative assumption regarding the potential release from this event.
- [[ ]](a)(4): There is not a specific EM within this topical report for this event to determine extent of fuel failure or confirm clad temperatures are in acceptable ranges (no release). For the PSAR, an event-specific calculation is performed and resides in the individual analysis. A mature methodology defining how to approach this scenario will be included in a future licensing document.
- [[ ]](a)(4): The quantification of the RWG leak is not a part of this EM. The Radiological Source Term Methodology [4] simply takes a conservative assumption regarding the potential release from this event.

The DBAs involving potential fuel failure and releases can be broadly categorized as:

- Fuel handling events
- Component failures and malfunctions
- Loss of cooling
- System leaks
- Sodium-water interaction
- Natural phenomena events.

The DBAs can also be grouped into two basic classes: in-vessel scenarios and ex-vessel scenarios.

**Table 2-1. Natrium DBAs with Radioactive Material Release.**

| Identifier | Topic | Summary |
|---|---|---|
| [[ | Core Blockage and Local Faults (DBA) | During at-power operations, the postulated initiating event is a blockage of fuel subchannels or other localized faults within the reactor core. While a manual shutdown would normally be initiated due to exceeding failed fuel limits, the reactor is assumed to continue operating at full power. The creep failure of all pins at the highest burnup is assumed for the affected single assembly using at power conditions. This single assembly failure does not have a significant impact on monitored safety related plant parameters. The vessel head is able to contain the radionuclide release (e.g., no pre-existing leak or seal failures are assumed). |
| | Excessive Sodium-Water Reaction in the PIC (DBA) | During ex-vessel fuel handling operations, the postulated initiating event is an excessive sodium water reaction in the PIC. The cladding integrity is failed following the sodium water reaction that occurs in the PIC. The BLTC boundary successfully retains the radionuclide release following the sodium water reaction (PIC boundary is not credited for DBA). The fuel cladding is failed within this scenario and radionuclide release occurs. |
| | Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel (DBA) | During refueling operations, the postulated initiating event is a fuel handling event while moving fuel in the reactor vessel. Assembly(s) impacted by the dropped component are damaged and a radionuclide release occurs. The functional containment barriers successfully retain the radionuclide release. |
| | Fuel Handling Event Occurs While Moving Fuel Assembly in the Spent Fuel Pool (SFP) (DBA) | During ex-vessel fuel handling operations, the postulated initiating event is a fuel handling event while moving fuel in the SFP. The fuel assembly is damaged and a radionuclide release occurs. The fuel cladding is failed within this scenario and radionuclide release occurs. |
| | Loss of EVST Cooling While Storing Fuel Assembly (DBA) | A loss of active EVST cooling occurs while handling spent fuel in the EVST. Analysis demonstrates 72 hours adiabatic heat up of the EVST and EVST vault to maintain fuel within performance limits. Longer term degraded heat removal conditions require further assessment. |
| ]][(a)(4)] | SCG leak Downstream the SCG Cell (DBA) | A Sodium Cover Gas System (SCG) leak occurs downstream the SCG cell. The Reactor Building (RXB) superstructure cannot contain radionuclides. The fuel cladding is not failed within this scenario, however radionuclide release occurs. |

| Identifier | Topic | Summary |
|---|---|---|
| [[ | SPS-I leak at the cold trap (DBA) | An Intermediate Sodium Processing System (SPS) leak occurs at the cold trap which is skid-mounted on the ground floor of the Reactor Auxiliary Building (RAB). A release of precipitated tritium from the SPS-I cold trap occurs. The RAB cannot contain radionuclides. The fuel cladding is not failed within this scenario, however radionuclide release occurs. |
| | RWG leak (DBA) | A Gaseous Rad Waste Processing system (RWG) leak occurs. The fuel cladding is not challenged within this scenario; however, radionuclide release occurs. |
| | SPS-P System leak in the RAB (DBA) | The SPS-P system leaks within the RAB. The SPS pump trips on low primary sodium level which stops the leakage due to system configuration. The SPS cell does not contain the radionuclides. The fuel cladding is not failed within this scenario, however radionuclide release occurs |
| ]](a)(4) | SPS-P Leak in the RXB (DBA) | The SPS-P system leaks inboard of the SPS isolation valves within the HAA. The SPS pump trips on low primary sodium level which stops the leakage due to system configuration. The HAA does not contain the radionuclides. The fuel cladding is not failed within this scenario, however radionuclide release occurs. |

# 3 ASSUMPTIONS REQUIRING VERIFICATION

## 3.1 Assumptions

The following assumptions are discussed in more detail in the individual EMs within Section 5 of this report. They are summarized here for information and provide context for items which are assumed to define the scope of an EM, determine conservative boundaries, or to identify areas in which future work is planned.

| Assumption Number | Description |
|---|---|
| 3.1 | Event and accident scenarios will be limited to DBAs with potential release. |
| 3.2 | This report is based on the current Natrium reactor system design and will be revised as appropriate as the reactor design and possible event scenarios mature. |
| 3.3 | The system leakage scenarios (DBAs resulting from leakage or breaks in the SPS, IHT, RWG, or SCG) are assumed during normal operation and not as part of, or consequence of, a different event. |
| 3.4 | It is presumed that [[ |
| 3.5 | ]](a)(4) |
| 3.6 | For a conservative scoping calculation of the potential radiological release happening during a fuel handling accident, a fuel assembly [[ ]](a)(4) |
| 3.7 | Detailed analysis of fuel drop accident considers limited scenarios [[ ]](a)(4) result in the worst possible fuel damage and the highest radiological release. |
| 3.8 | The partial flow blockage analysis is performed in a Natrium assembly that is operating at the fuel design limits. The assembly operates with a peak Linear Heat Generation Rate (LHGR) [[ ]](a)(4),ECI which may be updated with evolving fuel performance analysis, and a peak cladding temperature (PCT) [[ ]]((a)(4),ECI |

| Assumption Number | Description |
|---|---|
| 3.9 | The steady state tools used to design the reactor core have the fidelity to model the misloaded core [[ ]]$^{(a)(4)}$ |
| 3.10 | The final Natrium design [[ |
| 3.11 | ]]$^{(a)(4)}$ |
| 3.12 | The assembly to be misloaded is [[ ]]$^{(a)(4)}$ |
| 3.13 | The In-Vessel Transients with Radiological Release Methodology assumes that only Type 1 fuel is used. |

# 4 EVALUATION MODEL DEVELOPMENT AND ASSESSMENT

## 4.1 Background

The EMs considered in this report support the analysis of DBAs in the Natrium design involving clad or fuel failure with potential release of radionuclides into the coolant or beyond functional containment barriers and subsequent discharge into the environment. These EMs should be able to describe important phenomena identified by the relevant Phenomena Identification and Ranking Table (PIRT) studies with adequate accuracy and fidelity.

Table 4-1 provides the Figures of Merit (FOM) that have been used in the analyses of in-vessel DBAs without radiological release [3] and are also relevant to the in-vessel DBAs involving radiological release. Note that this listing represents the FOMs as a snapshot in time and will be updated accordingly as the FOMs evolve.

**Table 4-1. Figures of Merit for In-Vessel DBAs** [3]

| 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 centerline temperature. |
| Coolant temperature | High coolant temperature may cause sodium boiling in the reactor core, [[ ]](a)(4) 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 PCT 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 TATNF screening criteria incorporate cladding wastage and thermal creep criteria in assessing potential failure.<br><br>The TATNF FOM is constrained by the following:<br><br>• [[ ]](a)(4) |

| Figure of Merit | Descriptions and Significance |
|---|---|
| | • [[ ]](a)(4),ECI |

The full scope of Natrium EMs is composed of many codes and methods which span the range of initiating events that can result in clad or fuel failures and system leaks that lead to radiological release.

Figure 4.1-1 provides a high-level depiction of the EM workflow associated with DBAs. This high-level view illustrates the use of multiple independently licensed EMs to evaluate the dose consequences of Natrium DBA events. These individual EMs provide the foundational development and validation for the events described in this report. The EMs include:

- Core Design and Thermal Hydraulics [5],
- Design Basis Accident Methodology for In-Vessel Events without Radiological Release [3],
- Fuel and Control Assembly Qualification [6],
- Partial Flow Blockage Methodology [7]
- Radiological Source Term Methodology [4]. and
- Radiological Release Consequences Methodology [8]

This viewpoint aids in identifying possible gaps between the development, qualification and licensing of the individual EMs and their application to events resulting in a radiological release.

Some DBAs that have a potential for fuel failures will be in-vessel transients [[ ]](a)(4) These transients are analyzed with a multi-step process as discussed in Section 2.4 and illustrated in the upper path in Figure 4.1-1.

Using the TATNF screening criteria effectively identifies the bounding events and filters out those that do not require more detailed analysis. [[ ]](a)(4) where fuel pin failure is possible due to a combination of factors including prior irradiation exposure, the extent of cladding wastage (fuel-clad chemical interaction, eutectic, etc.), and the extent of cladding mechanical creep. The TATNF screening criteria provides a framework to decide if a safety analysis event that occurs in this temperature range requires further assessment as described in Section 5.1.

Other DBA events that can lead to radiological releases are presented in the lower branch of Figure 4.1-1 and discussed in Sections 5.2 through Section 5.5.

Regardless of the methods used to determine the clad and/or fuel pin failures, all events will provide information to the radiological source term EM [4] and then to the dose consequence EM to determine the transport and consequence of radiological release to the environment [8].

(a)(4)

**Figure 4.1-1. EM Calculational Devices and Analysis Workflow.**

## 4.2 Evaluation Model Development

As shown in Table 4-2, the Natrium events with potential fuel failure and/or radiological release can be grouped based on the common important phenomena, location, and modeling objectives/requirements. Suitable modeling strategies and EM are then established for each group.

**Table 4-2. Representative Events with Potential Fuel Failure and Radiological Release.**

| Event Category | Event | Location | Phenomena | Suitable Software |
|---|---|---|---|---|
| Core/PHT/IHT events | Core symmetric events | In-vessel | Core neutronics, fuel behavior, and coolant thermal hydraulics | [[ ]](a)(4) |
| | Core asymmetric events - one-pump trip | In-vessel | Core neutronics, fuel behavior, and coolant thermal hydraulics | |
| | Loss of heat sink, RAC long-term transient | Ex-vessel | Heat removal from PHT system | |

| | | | | |
|---|---|---|---|---|
| Local events | Partial flow blockage | In-vessel | Thermal hydraulics behavior/Release/Transport | [[ |
| | Fuel misload | In-vessel | Coupled thermal hydraulics/neutronics/fuel performance behavior | |
| Fuel handling accidents | In-vessel fuel drop, ex-vessel fuel drop at different locations | In-/Ex-vessel | Transport and consequence of radiological release | |
| | | | Thermo-mechanical/structural-mechanical behavior/failure | |
| Sodium/gas leaks/releases | Sodium/gas leaks/releases | Ex-vessel | Transport of quantified radionuclide release | ]](a)(4) |

[[ ]](a)(4)

Accident and safety analysis of SFRs is not as mature as that of light water reactors and many gaps in the SFR safety modeling capabilities have been identified and continue to be addressed. [9]

Instead of developing new modeling capabilities, safety analysis of the Natrium design is primarily focused on the use of readily available modeling tools which are selected and acquired via TerraPower's Acquired Software Quality Assurance Plan under Safety Analysis and Risk. The plan provides a process framework supporting the quality assurance (QA) requirements for software (computer codes) that perform safety-related or non-safety-related analysis in the Natrium plant, [[ ]](a)(4) In addition, the gap analysis and planned maturation activities for each potential software are discussed. Specific sections are included in the plan to discuss the commercial grade dedication (CGD) that will be implemented for commercially acquired software that will be used for safety-related applications.

DBAs involving reactor core and PHT systems and components [[ ]](a)(4) models the core at the assembly level, i.e., each fuel assembly is represented by a single channel comprising the fuel, cladding, coolant, and associated structure; detailed analysis at the local fuel pin/subchannel level requires the ability to model the multidimensional phenomena within the fuel assembly. The codes selected and developed for this use are [[ ]](a)(4) which characterize the individual fuel pins that are expected to fail. The failed rod(s) initial radionuclide inventory is used in the radiological source term method [4] that determines the leakage through facility systems and the dose consequences of the release to the environment.

In-vessel partial flow blockage events (Section 5.2) that involve subchannel coolant thermal hydraulics [[ ]](a)(4)

Other In-vessel events that result in pin failures are Fuel Misloads (Section 5.3), and Fuel Handling Events (Section 5.4). Fuel Handling Events may also occur ex-vessel in the Ex-Vessel Fuel Handling Machine (EVHM), EVST, the PIC, the SFP, and possibly during transport between the locations.

Structural analysis software, [[ ]](a)(4) can be used to analyze the thermo-mechanical and structural mechanical behavior of a fuel assembly and evaluate potential failure resulting from fuel handling accidents.

DBAs involving system leaks are often analyzed with use of the radiological source term method [4], [[ ]](a)(4) as described in Section 5.5. However, some system leaks such as large leaks in the IHT could lead to a sequence of events that lead to an In-Vessel Transient of sufficient magnitude to cause fuel pin failures using the In-Vessel Transient Methodology (see Section 5.1).

If an event involves more than one of the phenomena mentioned above, different modeling tools can be used together with different code coupling/interfacing strategies (one-way or two-way) to be employed. [[

]](a)(4)

## 4.3 Phenomena Identification and Ranking Tables (PIRT)

Important phenomena and processes are identified and ranked for each event category via the PIRT study. The PIRT objective is to identify safety-relevant phenomena and processes for the considered event, rank their importance based on pre-established FOMs, and rank the status of knowledge to build a technical basis to develop the EM.

The process for establishing a PIRT is iterative in nature and follows a pattern of progressive elaboration that consistently drives the PIRT to move from qualitative discussion to quantitative descriptions. Whereas early phases of the PIRT process make heavy use of independent expert opinion and precedent PIRTs where applicable, the later phases take benefit of detailed computational analyses that provide direct and indirect evidence of phenomenological importance and impact of identified items. This quantified experience is key to ensure the credibility of the finished PIRT, where analytic predictions clearly show the importance (or lack of) for each PIRT item over the entire domain of application.

More details on PIRTs for DBA events with potential fuel failure and radiological release are documented in later sections of this report. In particular, the following PIRT reports are referenced:

- Phenomena Identification and Ranking Table Report for Natrium Other Quantified Events (Section 5.1, Table 4-3)
- Phenomena Identification and Ranking Table Report for Natrium Partial Flow Blockage within a Subassembly Evaluation Model (Section 5.2)
- Natrium Topical Report: Fuel and Control Assembly Qualification [6] (Table 4-4)

The PIRT for Natrium Other Quantified Events (OQEs) identifies important phenomena associated with [[ ]](a)(4) the PIRT for partial flow blockage is described in TP-LIC-RPT-0008 Rev. 0, “Partial Flow Blockage Methodology” [7]; and Table 6-3 in NAT-2806 Rev. 0, “Natrium Topical Report: Fuel and Control Assembly Qualification” [6] summarizes the high-important phenomena associated with fuel and absorber pin behavior.

The PIRTs presented in this report take into account important process and phenomena rankings described in the PIRTs available for in-vessel DBAs without radiological release [3], LBEs without fuel failure, and radiological source term events.

## 4.4 Evaluation Model Assessment

TerraPower's Quality Assurance Program Description (QAPD) [10] and Software Management Procedure (SMP) detail the QA requirements and processes. The QAPD and SMP comply with the applicable requirements of ASME NQA-1-2015 [11], 10 CFR Part 50 Appendix B, and RG 1.28 [12]. TerraPower utilizes the graded approach by implementing the existing QA program controls for software that performs safety-related and/or non-safety-related applications.

The adequacy assessment of the EMs for DBAs with potential fuel failure and radiological release is guided by the TerraPower's Acquired Software Quality Assurance Plan under Safety Analysis and Risk. For the codes to be accepted for safety-related applications, they should be assessed based on the list of legacy verification and validation (V&V) activities including verification test suite cases, legacy validations of severe accident modules, and benchmark activities. The assessment also identifies the verification, validation, and uncertainty quantification gaps that require closure. Some codes are still under further development [[ ]](a)(4) and plans for the code maturation activities have been established.

The first step in the model assessment is to investigate the availability of legacy experimental data and evaluate the pedigree of the data. An Assessment Matrix is created for each methodology described herein that has an associated PIRT. The fuel failure phenomena identified in the PIRT as High and Medium importance are matched against the available experimental data. These fuel failure phenomena are generally considered to be applicable to all events with fuel failure, but in practice will likely be modeled for in-vessel events using [[ ]](a)(4). Available experimental data is the historical data in the applied technology reports and journal papers. Based on the Assessment Matrix, testing needs will be identified.

Table 4-3 and Table 4-4 show the assessment matrices for the fuel failure phenomena. Note that the phenomena listed in Table 4-3 was developed for OQEs but is bounding and applicable for LBEs and DBAs with potential fuel failure and radiological release, and identifies important phenomena associated with [[ ]](a)(4)

For in-vessel events with potential fuel failure, given the overlap of important processes and phenomena in DBAs with and without fuel failure, the assessment matrices discussed here mostly focus on fuel failure phenomena. More detailed PIRT and assessment matrix for in-vessel DBAs without fuel failure can be found in NAT-9390 [3].

A partial flow blockage assessment matrix is included in the Partial Flow Blockage Methodology report documented in TP-LIC-RPT-0008 [7]. It should be noted that partial flow blockage phenomena does not include any fuel failure phenomena since the methodology only covers up to the fuel failure point.

For the Fuel Misload Methodology, the code qualification, verification, and validation in NAT-2806 [6] and TP-LIC-RPT-0011 [5] are leveraged for core design, fuel performance, and thermal hydraulic codes. Fuel misloads do not introduce phenomena beyond those normally modeled in steady state analysis, thus, the code qualification, verification, and validation utilized for steady state core design, thermal hydraulics, and fuel performance are serving the function of an assessment matrix for the Fuel Misload Methodology.

For the Fuel Handling Accident Methodology and Sodium Liquid and Gas Leak Methodology, assessment matrices were developed as part of the Source Term EM (NAT-9392 [4]). However, it is noted that the phenomena associated with these assessments are related to the release and transport of radionuclides, and not to the dynamics and structural analysis that is the subject of the Fuel Handling Accident Methodology in this report, nor the calculation of the leak rate and timing that is the subject of the Sodium Liquid and Gas Leak Methodology in this report. Assessment matrices have not been developed for these parts of the methodology at this time but may be developed in the future as those methods are developed and matured.

Assessment matrices for Loss of Active Cooling and Excessive Sodium-Water reaction have not been developed at this time, but may be developed in the future as those methods are developed and matured.

**Table 4-3. Assessment Matrix for High/Medium Importance Fuel Failure Phenomena in OQEs.**

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]](a)(4)

**Table 4-4. Assessment Matrix for High-Importance Fuel Failure Phenomena.**

| High-Importance Phenomena | Applicable Design Limit | Overview of Testing[2] |
|---|---|---|
| [[ | | |
| | | |
| | | |

]](a)(4)

[2] Note that some of these identified tests may be eliminated pending additional analysis or retrieval of additional historic data.

[3] [[

]](a)(4)

[[

| High-Importance Phenomena | Applicable Design Limit | Overview of Testing[2] |
|---|---|---|
| | | |
| | | |

]](a)(4)

After the Assessment Matrix is created, data will be acquired from the identified resources. After the data is acquired, it will be qualified based on the TerraPower's existing data qualification procedure.

An initial assessment database has been constructed and is shown in Appendix B.

# 5 EVENT-SPECIFIC METHODOLOGY

## 5.1 In-vessel Transients with Radiological Release Methodology

### 5.1.1 Purpose and Scope

This methodology [[ ]](a)(4) to the analysis of DBAs that lead to clad and fuel failures, and which result in the release of radionuclides into the coolant. There is a wide variation of initiating events and event scenarios in the DBA event class [[

]](a)(4) Therefore, the boundary conditions from the [[ ]](a)(4) event simulations performed following NAT-9390, "Design Basis Accident Methodology for In-Vessel Events without Radiological Release" [3] and as augmented by the In-Vessel DBAs and Non-DBA LBEs without Radiological Release Application Methods will remain applicable.

As illustrated in Figure 4.1-1, [[ ]](a)(4) event simulation is performed to determine the extent of the challenge to fuel pin cladding integrity. The first step in this evaluation assesses the margin to the conservative TATNF screening criteria, which incorporates cladding wastage and thermal creep criteria in assessing potential failure.

The following filtering criteria are used to determine potential fuel failure. The most limiting channels [[ ]](a)(4) are identified based on the following screening criteria:

1) [[

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

If TATNF is not violated for a given transient, then no radiological release occurs. However, TATNF incorporates a conservative approach to fuel performance modeling which bound a wide variety of potential temperature histories. [[

]]$^{(a)(4)}$ passed to the DSAW described in Section 5.1.5.

DSAW provides a more mechanistic, event specific approach to performance analysis. [[
]]$^{(a)(4)}$ Additionally, for slower transients, [[
]]$^{(a)(4)}$ used by TATNF.

The present version of DSAW does not allow [[

]]$^{(a)(4)}$ that integrates the severe accident modules as described in the following sections is planned to be employed.

### 5.1.2 Assumptions

- Event and accident scenarios will be limited to DBAs with fuel failures and radiological release.
- The plan is based on the current Natrium reactor system design and will be revised as appropriate as the reactor design and possible event scenarios mature.
- The In-Vessel Transients with Radiological Release Methodology assumes that only Type 1 fuel is used.

### 5.1.3 EM Scope and Requirements

Up to the onset of cladding and/or fuel failures the [[ ]]$^{(a)(4)}$ DBA without release is identical to the method presented here. Therefore, the entire DBA modeling approach, PIRT phenomena and uncertainties established for the In-Vessel DBAs without Release [3] Methodology will be applied to this EM, subject to confirmation that models and uncertainties remain applicable for the range of conditions exhibited in the limiting DBA scenarios.

Three event phases are identified associated with severe accidents:

Initiating: This stage defines phenomena for an event as it transitions from an accident to a severe accident. [[

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

Transition: [[

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

Termination: [[

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

As stated in Section 2.1, DBAs only credit SR SSCs to demonstrate compliance with the 10 CFR 50.34 dose limits. If a DBA does exceed the dose limits, then new SR SSCs are selected, and their required safety functions defined until the dose limits are met. Some DBAs will experience insufficient heat removal from the fuel, [[

]]$^{(a)(4)}$ The range of phenomena associated with in-vessel transient DBAs would not transition to a severe accident as described for the Initiating phase.

While DBAs will not exhibit fuel failure phenomena (e.g., coolant boiling or fuel melting) associated with the most severe BDBE events or OQEs, the evaluation model described in this section is being developed to address the full scope of DBA, BDBE and OQEs. A PIRT has been established for the ULOF, ULOHS and Unprotected Transient Over-Power (UTOP) by internal and external panelists and documented in detail in the PIRT for Natrium OQE. While these three events do not necessarily consider the characteristics of all the possible BDBEs, they were considered adequate to identify phenomena that may be expected for the more frequent BDBEs and are applicable and bounding for in-vessel DBAs.

The Natrium OQE PIRT was supported by scoping calculations for the ULOF, ULOHS, and UTOP transients and included in Appendix A of the PIRT document. The calculations were performed for Beginning of Equilibrium Cycle core condition of each scenario and were extended to investigate the beyond BDBE consequences [[

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

In general, the results show that flow reduction or reactivity insertion without scram leads to a rapid core heat up, [[

]]$^{(a)(4)}$ This is driven by negative reactivity feedback from Doppler, fuel axial expansion, core radial expansion, control rod driveline expansion, and coolant density. Then the negative reactivity results in a monotonic decrease in core power level.

The sensitivity studies show that a highly conservative transient initiator leads to a substantial increase in core temperatures, resulting in coolant boiling, fuel melting, fuel relocation and ejection into the coolant channel with cladding breach. In the sensitivity evaluation [[ ]](a)(4) would lead to the onset of local boiling and fuel/cladding failures.

Combined PIRT results for ULOF, ULOHS and UTOP events with High/Medium importance ranked phenomena are tabulated in Table 5-1. This reflects the highest importance with the associated lowest state of knowledge for each phenomenon. Note that Table 5-1 does not include the severe accident phenomena associated with Stage 2 of the ULOHS from the Natrium OQE PIRT as they are beyond the scope of this report. Details on the rationale and rankings for the importance and knowledge level for individual phenomena and processes for each event analyzed are given in the PIRT report.

**Table 5-1. Combined PIRT for ULOF, ULOHS and UTOP LBEs with Radiological Release with High/Medium Importance Phenomena.**

[[

| No. | Phenomenon | Importance Ranking | State of Knowledge |
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]](a)(4)

### 5.1.4 EM Description

The In-Vessel Transient with Radiological Release EM is an extension to the EM established for In-Vessel DBAs without Release [3]. The extension includes the use [[ ]](a)(4) severe accident modules to analyze fuel pin failures and subsequent relocation. As such, this EM incorporates all the qualification, verification and validation associated with Reference [3] [[ ]](a)(4)

It should be noted that two independent fuel performance models [[ ]](a)(4) have been matured for the purpose of predicting fuel pin damage and failure for use in safety analysis methodologies [6] and both codes will be verified and validated for use within the EM. These two models were developed independently, utilize different numerical methods, and differ in the approaches to modeling certain phenomena. The [[ ]] (a)(4) model is used by the DSAW to conservatively determine the peak-pin margin to failure (see Section 5.1.5). If the DSAW results demonstrate that assembly wide fuel failures are expected, the [[

]](a)(4) the following modules are required to address cladding and/or fuel failures that occur during the in-vessel transient simulation.

[[

]](a)(4)

[[

]](a)(4)

### 5.1.5 In-Vessel Transient Evaluation Workflow

(a)(4)

**Figure 5.1-1. DSAW Data Flow.**

The DSAW for core transient analysis (Figure 5.1-1) [[

]](a)(4) Natrium safety analysis is provided in Appendix A9.1.2. [[

]](a)(4)

The following computer programs are used in the DSAW.

[[

]](a)(4)

[[

]](a)(4)

### 5.1.6 EM Assessment

The adequacy assessment of the EM for DBAs with release, [[ ]](a)(4) is guided by the TerraPower's Acquired Software Quality Assurance Plan under Safety Analysis and Risk. [[

]](a)(4) provides a summary list of legacy V&V activities including verification test suite cases, legacy validations of severe accident modules, and benchmark activities. It also identifies the verification, validation, and uncertainty quantification gaps that require closure.

[[ ]](a)(4) models severe core disruption accidents with coolant boiling and fuel melting and relocation, is less developed than the remaining part which analyzes the thermal-hydraulic processes in other plant systems and components outside the reactor core. The fuel performance and failure analysis [[ ]](a)(4) software quality assurance over its entire lifetime. The implementation of an SQA program for [[

]](a)(4) the fuel performance and fuel failure analysis part of the code [[ ]](a)(4) Ongoing work is planned to be complete prior to TerraPower's submittal of an operating license application, and information on [[ ]](a)(4) to fill the quality gap to complete the CGD at TerraPower will be included in a future licensing submittal.

[[ ]](a)(4) maturation activities at TerraPower has been established per TerraPower's Acquired Software Quality Assurance Plan under Safety Analysis and Risk. The code assessment including V&V is included in this plan. [[ ]](a)(4) analyze several unprotected events (ULOF, ULOHS, UTOP, etc.), that potentially involve fuel failure, in support of the PIRT process for LBEs with release and OQEs. [[ ]](a)(4) has been integrally validated in a study of the ULOF accident with cladding/fuel failure in a SFR using the CABRI integral effect test (IET) data [14]. Ongoing work in this area is planned to be complete prior to TerraPower's submittal of an operating license application, and that information will be included in a future licensing submittal.

The DBAs that lead to potential fuel failures are associated with events that likely result in a significant heat up of the PHT with either symmetric or asymmetric boundary conditions at the inlet of the core. The current level of fidelity [[ ]](a)(4) cannot resolve the multidimensional processes that take place in the large pool sections of the

PHT system for asymmetric events or the dynamic impact of thermal stratification in the warm and hot pools. While the loss of two IHX clearly bounds the loss of a single IHX, DBA evaluations performed to date of the Natrium design show that the [[ ]](a)(4) result in cladding failures. However, the current level of fidelity [[ ]](a)(4) cannot resolve the multidimensional processes that take place in the large pool sections of the PHT system for asymmetric events or the dynamic impact of thermal stratification in the warm and hot pools A longer term development project to support the FSAR is the development and qualification of an Integrated Pool Methodology. This methodology will leverage [[ ]](a)(4) to identify and address non-conservatisms [[ ]](a)(4) has been endorsed by the NRC to address complex issues in Light Water Reactor (LWR) licensing when combined with appropriate experimental data [15]. [[ ]](a)(4) Ongoing work in this area is planned to be complete prior to TerraPower's submittal of an operating license application, and that information will be included in a future licensing submittal.

## 5.2 Partial Flow Blockage Methodology

Partial blockage of the coolant flow in a fuel assembly has been considered as one of the important safety issues of SFRs. It is characterized by the tight spacing of fuel pins, high power density and high burnup fuel. Partial flow blockage may be initiated due to the accumulation of debris circulated in the primary sodium, failure of wire-wrapped spacers, and from swelling or bowing of the fuel pins. The partial flow blockage can cause the temperature rise in the wake region behind the blockage; therefore, it may lead to the potential for sodium boiling, dry out, cladding thermal failure and fuel melting.

Full discussion of this method and of the work that is ongoing in these areas is captured in TP-LIC-RPT-0008 Rev. 0, "Partial Flow Blockage Methodology [7]. A summary discussion of the EM that has been developed is provided below to provide context within the scope of DBAs with radiological release.

### 5.2.1 Purpose and Scope

The purpose of the partial flow blockage analysis is to demonstrate that the Natrium design satisfies the regulatory requirements of dose consequences for DBAs with Release Methodology with enough safety margins and meets CP and Operating License guidelines. This goal is achieved by confirming in the analyses that the system responses to DBAs with partial flow blockage within a fuel assembly satisfy all relevant acceptance criteria during the normal operating conditions.

The safety objective of the flow blockage analysis is to investigate the potential effects of partial flow blockage within a Natrium fuel assembly on fuel integrity based on the PCT. The fuel integrity can be maintained if the cladding damage is avoided. Rods that exceed the PCT steady state acceptance criteria are treated as failed.

The scope of this analysis is to provide bounding cladding temperatures for an infinitely thin, fully impermeable blockage within a Natrium assembly at steady state operating conditions.

### 5.2.2 Assumptions

Assumptions are discussed in detail in the Partial Flow Blockage Methodology topical report [7]. [[

]](a)(4)

### 5.2.3 Acceptance Criteria

PCT is used in partial flow blockage analysis as the acceptance criteria.

### 5.2.4 EM Scope and Requirements

The scope and requirements for partial flow blockage EM are established via the PIRT process. Details of this PIRT process can be found in Partial Flow Blockage Methodology topical report [7].

### 5.2.5 EM Description

The safety analysis of partial flow blockage is performed with respect to the frequency-based criteria. The EM provides a bounding temperature for an infinitely thin, fully impermeable blockage within a Natrium assembly at steady state operating conditions. The blockage sizes are selected per a frequency-based criteria. This event is undetectable prior to fuel failure. This hypothetical planar blockage bounds the following credible events: collapsed wire wrap, rod bowing without contact and lodged foreign material. EM includes the upper bound for the maximum PCT, the number of fuel pins, and the potential associated radiological release. The thermal hydraulic analysis of partial flow blockage is performed using [[ ]](a)(4) the semi-empirical model which is in the process of being validated against historical ORNL data for central 6 subchannel blockages and 14 subchannel edge blockages [16]. [[ ]](a)(4)

Additional detail describing the EM is available in the Partial Flow Blockage Methodology topical report [7].

[[

]](a)(4)

[[

]](a)(4)

### 5.2.6 EM Assessment

A full discussion of this method is captured in the Partial Flow Blockage Methodology topical report [7]. As such, this report refers to the partial flow blockage report for the qualification, verification, and validation plans associated with the EM summarized here in Section 5.2. Ongoing work in this area is planned to be complete prior to TerraPower's submittal of an operating license application, and that information will be included in a future licensing submittal.

## 5.3 Fuel Misload Methodology

### 5.3.1 Purpose and Scope

During refueling outages, fuel assemblies are discharged or shuffled to new core locations and fresh fuel is loaded. The purpose of this methodology is to analyze the consequences of moving an assembly to the wrong core location or loading it in the right location but the wrong orientation.

The Natrium core has two main enrichment zones (inner and outer), with the outer zone being higher enrichment to flatten power. [[

]](a)(4) Figure 5.3-1 shows the core layout, and the orange and green arrows indicate the convergent shuffle direction of the fuel assemblies.

[[ ]](a)(4)

**Figure 5.3-1. Equilibrium Core Fuel Layout.**

### 5.3.2 Assumptions

- The steady state tools used to design the reactor core have the fidelity to model the misloaded core [[ ]](a)(4)
- The final Natrium design [[ ]](a)(4)
- [[ ]](a)(4)
- The assembly to be misloaded is [[ ]](a)(4)

### 5.3.3 EM Scope and Requirements

The fuel misload has one dominant highly ranked phenomenon, which is the change in the local power distribution. The core power distribution drives the core temperature distribution at steady-state conditions. The power distribution is a function of the core composition, burnup distribution, and geometry. The magnitude of the change to the local power distribution for the misload event depends on the change of the pin-level isotopic distribution between the intended and misloaded core configuration. [[ ]](a)(4)

The processes and phenomena described above are modeled with the EM described in the following subsection.

### 5.3.4 EM Description

The Fuel Misload Event is analyzed using [[ ]](a)(4)

For PSAR evaluations, 10 cases were selected covering a sample of misloaded fresh, once burned, twice burned, and thrice burned assemblies at beginning-of-life and beginning of equilibrium core conditions. [[ ]](a)(4)

[[

]]^(a)(4)

The methodology for determining the limiting assembly and fuel pin during the fuel misload transients has not been finalized and future work may change the position and number of assemblies involved in these cases. Ongoing work in this area is planned to be complete prior to TerraPower's submittal of an operating license application, and that information will be included in a future licensing submittal.

#### 5.3.5 EM Assessment

Additional detail on the core design and thermal hydraulic codes used to predict the steady-state local power and temperature distributions is provided in the fuel qualification topical report [6] and in TP-LIC-RPT-0011 Rev. 0 “Core Nuclear and Thermal Hydraulic Design Technical Report” [5].

As such, this report refers to the code qualification, verification, and validation plans included in those reports. Ongoing work in this area is planned to be complete prior to TerraPower's submittal of an operating license application, and that information will be included in a future licensing submittal.

### 5.4 Fuel Handling Accident Methodology

#### 5.4.1 Purpose and Scope

Fuel assemblies can be damaged in various fuel handling (FH) events during (i) insertion or removal from reactor core, (ii) in-vessel fuel assembly movement, (iii) ex-vessel fuel assembly movement between the EVST and washing station, or due to (iv) inadvertent action causing spent fuel assembly crush, (v) fuel assembly or loaded fuel cask drop in spent fuel pool.

These events need to be analyzed to determine the possible damage and final configurations of both the dropped and impacted fuel assemblies. In addition, the potential release of radionuclides resulted from such FH accidents as well as their leakage to the environment needs to be quantified.

In this section, only the EM for analysis of structural-mechanical behavior and failure of dropped/impacted fuel assembly is discussed. The transport and consequence of radiological release resulted from a DBA FH event are analyzed by the Radiological Source Term Methodology [4].

#### 5.4.2 Assumptions

- [[
]]^(a)(4)
- For a conservative scoping calculation of the potential radiological release happening during a fuel handling accident, [[
]]^(a)(4)

- Detailed analysis of a fuel drop accident considers limiting scenarios [[ ]](a)(4) result in the worst possible fuel damage and the highest radiological release.

### 5.4.3 Acceptance criteria

The fuel drop analysis used for the evaluation of LBEs and DBAs may be performed as part of analysis for other design purposes. As a result, the following acceptance criteria may be defined generally for fuel drop analysis:

- Dropped fuel assemblies must not result in fuel cladding mechanical failure in either the dropped assembly or any targeted structures.
- Dropped fuel assemblies must not create unacceptable core component conditions that would impact safe reactor operations (e.g., local criticality, loose parts within the components, reduced flow through the components, etc.).

The methodology established in this section, however, only covers the analysis of mechanical damage of the dropped assembly and not the consequence of potential radiological release resulted from such a mechanical failure. Analysis of the consequence of potential radiological release resulted from a FH accident involves other acceptance criteria pertaining to the source term analysis which is documented in [4]. As such, neither of the acceptance criteria listed above apply to this methodology, and there are not necessarily acceptance criteria for this EM. Rather, it is used to provide fuel failure information to the downstream source term analysis, and failure of the fuel is acceptable as long as the ultimate dose consequences acceptance criteria are met.

### 5.4.4 EM Scope and Requirements

A fuel handling accident can be initiated by FH machine malfunction and/or operator errors. [[ ]](a)(4) However, some mechanisms potentially causing mechanical damage to fuel assemblies during a fuel drop event can be identified as follows.

- Stress, strain, and loading limits of the fuel assembly components including assembly duct, fuel rods, spacers, receptacle, etc.
- Fatigue of structural components resulted from cyclic and dynamic load histories.
- Elastic/inelastic behavior (deformation) of components under loading
- Mechanical fracturing caused by dynamic loads or impact of the dropped assembly on other structure(s).

The EM to be used for analysis of fuel assembly mechanical failure during a FH accident should be able to model the above-mentioned processes and phenomena.

A detailed PIRT for radiological release and consequence resulted from FH accidents can be found in Table 2 of the Radiological Source Term Methodology [4].

### 5.4.5 EM Description

The dynamic structural behavior and integrity of fuel assemblies subjected to mechanical impact in a FH event can be analyzed in detail with the (nonlinear) Finite Element Analysis (FEA) software, [[ ]](a)(4) The FEA software can provide predictions of the fuel assembly mechanical stress and impact force for different fuel drop scenarios defined by the [[ ]](a)(4) etc., which can then be compared with the fuel assembly strength limits to determine the extent of the fuel damage (number of damaged fuel pins). This result can then be used to quantify the radiological release for the event. The transport inside reactor containment and the potential release to the environment of radionuclides is then analyzed with [[ ]](a)(4) described in the Sodium and Gas Leak Methodology Section 5.5, and the Radiological Source Term Methodology [4].

The FEA software employs a finite element method , where the behavior of a solid or fluid system in two or three space dimensions is solved by subdividing a large computational domain into smaller, simpler parts called finite elements. This is achieved by a particular discretization in the space dimensions, which is implemented by the construction of a mesh or a computational grid of the analyzed domain.

A finite element model of a Natrium fuel assembly has been built and used for the preliminary analysis of core assembly drop accidents. The method requires inputs of geometric properties (such as fuel pins, assembly duct, receptacle, spacers, etc.) and material properties of the core assembly, the stiffness of the impacting receptacle or surface, the boundary conditions of the drop scenario, and some experimentally determined factors, such as the impact damping coefficient. The output of the method is impact load histories that may be used to perform stress analyses on core assemblies.

[[ ]](a)(4)

The EM used to analyze the transport and consequence of the FH DBA radiological release will be based on [[ ]](a)(4) which are described in detail in Section 5.5 and the Radiological Source Term Methodology [4].

### 5.4.6 EM Assessment

FEA software [[ ]](a)(4) have a very broad range of applicability in different industries, such as aerospace, automotive, machinery, oil & energy, etc., as they provide detailed insight and offer a unique tool for structural analysis. However, there are still limitations to their applicability as a routine tool for safety justification of nuclear power plants.

Assessment of the FEA software selected for analysis of Natrium safety problems is guided [[ ]](a)(4) Ongoing work in this area is planned to be complete prior to TerraPower's submittal of an operating license application, and that information will be included in a future licensing submittal.

[4] [[ ]](a)(4)

In performing the FEA for a fuel drop event, additional inaccuracies and uncertainties may arise due to:

(i) Geometrical/structural complexity of the fuel assembly which contains hundreds of fuel pins and many other supporting structural components; and

(ii) Difficulty in defining the [[ ]][(a)(4)]

The conservative definition of a fuel drop scenario can be used to obtain the maximum fuel damage in such an event.

## 5.5 Sodium Liquid and Gas Leak Methodology

### 5.5.1 Purpose and Scope

In addition to the transport of radionuclides (RNs) from fuel failures due to In-Vessel Transients, Fuel Handling events and Partial Flow Blockage through facility systems, transport of RNs released due to leaks were quantified as important phenomena relevant to DBAs with radiological material releases.

System leak scenarios include leaks associated with the Sodium Cover Gas System (SCG) [[ ]](a)(4) that has leaked from the fuel; the SPS which would include [[ ]](a)(4) and the IHT which would include [[ ]](a)(4)

The purpose of this EM is to determine and quantify leaks for dose calculations. The analysis includes the extent of leaks and releases based on the event initiation - the location, timing, system conditions, and propagation.

For PSAR, the Radiological Source Term Methodology approach taken for Sodium and Gas Leaks is to determine or assume a maximized release (e.g. complete system release, or all available volume prior to pump trip). A detailed methodology for mechanistically determining the specific leak conditions (e.g. mass and energy release) has not yet been developed. [[ ]](a)(4)

### 5.5.2 Assumptions

- The system leakage scenarios are assumed during normal operation and not as part of, or consequence of, a different event.
- [[ ]](a)(4)

### 5.5.3 EM Scope and Requirements

The EM established for analysis of sodium and gas leak events should have the capability to model important processes and phenomena detailed in the PIRT study in the Source Term topical report [4].

Table 5-2 summarizes the medium and highly ranked phenomena where knowledge level is equal or lower than the importance ranking.

**Table 5-2. List of Medium and High Importance phenomena in SPS leak events [4].**

[[

| No. | Phenomenon / Process | Description | Importance Ranking | Rationale for Importance Ranking | Knowledge Level | Rationale for Knowledge Level |
|---|---|---|---|---|---|---|
| | | | | | | |
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]](a)(4)

[[

| No. | Phenomenon / Process | Description | Importance Ranking | Rationale for Importance Ranking | Knowledge Level | Rationale for Knowledge Level |
|---|---|---|---|---|---|---|
| | | | | | | |
| | | | | | | |

]](a)(4)

### 5.5.4 EM Description

As described in the Radiological Source Term Methodology [4] the EM is comprised of the analysis codes: [[ ]](a)(4) and the output/input interfaces between each calculational device. Figure 5.5-1 provides the EM diagram for the SPS, SCG and IHT leak evaluations. The portions of the figure relevant to the DBA with release EM are the [[ ]](a)(4) of the "System Leak Rate," when needed.

[[ ]](a)(4)

**Figure 5.5-1. Sodium Cleanup System and IHT Leak EM Diagram.**

The SCG and the SPS source term will be based on the coolant inventory during normal operation. The steady-state inventory for the sodium cleanup system [[ ]](a)(4) The modeling of the potential leakage from the steady state normal core or primary system activity [[ ]](a)(4) The required information from upstream evaluations includes [[ ]](a)(4)

The sodium leak into the atmosphere itself, confined in a building or outside, likely will be evaluated [[ ]](a)(4) to account for any sodium reaction effects. A brief description of each of the analysis codes can be found in the Radiological Source Term Methodology [4].

### 5.5.5 EM Assessment

As noted in the Radiological Source Term Methodology [4], the EM acceptance assessment is planned to be performed for the following activities:

- Acceptance test plans for each individual code mentioned has formally been completed and effort will begin on the resolution of identified gaps [[ ]](a)(4) There are no known gaps [[ ]](a)(4) relevant to its use in the Source Term EM.

- Assessments of individual model fidelity, accuracy, and scaling for sources. As part of this task, it is anticipated that it will address the integrated calculations and consideration for data distortions.

As part of the EM process biases and uncertainties will be addressed for all LBEs with exception to the DBAs. The DBAs identified as part of the EM will use an approach that considers conservatisms.

The radiological source term EM activity will address prediction of FOMs through incorporation of biases and uncertainties into the various code mathematical models. The overall quantification of uncertainties will address each of the calculational devices as well as for the propagation of uncertainties through the series of codes used in the event evaluation.

As such, this report refers to the radiological source term report [4] for the qualification, verification, and validation plans associated with the EM discussed here in Section 5.5.

Ongoing work in this area is planned to be complete prior to TerraPower's submittal of an operating license application, and that information will be included in a future licensing submittal.

# 6 SUMMARY

## 6.1 Summary of Codes Selected

A wide range of methods and EMs established for analysis of DBAs with potential fuel failure and radiological release in the Natrium plant has been summarized in this report. The diversity of the events and phenomena involved necessitates different analysis approaches and EMs, ranging from conservative estimation to first-principle modeling. Even with first-principle modeling, conservative assumptions are often necessary in regards to the initial conditions, boundary conditions, and some model parameters. Justification for the conservatism or accuracy of the EM prediction affected by the specification of the initial/boundary conditions as well as the choice of model parameters is to be provided in each EM application. Overall prediction uncertainty also needs to be considered and quantified.

The list of the software includes:

- [[

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

Most of the above codes are acquired by TerraPower via the TerraPower Safety Software Gap Analysis, CGD, and Maturation plan guided by TerraPower's QAPD [10] and SMP. Work is ongoing for the codes to be accepted for Natrium safety-related applications, with assessments planned based on a list of legacy V&V activities including verification test suite cases, legacy validations of severe accident modules, and benchmark activities.

Additionally, the assessment identifies the verification, validation, and uncertainty quantification gaps that require closure.

[[ ]](a)(4) has been acquired with following assessments:

- Software Dedication Acceptance Test Plan
- Software Dedication Technical Evaluation Report
- Software Dedication Acceptance Test Report
- Software Dedication Report

Some codes are still under further development [[ ]](a)(4) and plans for the code maturation activities have been established. Assessments of important closure models and integrated performance of the EMs are planned together with the acquisition of relevant experimental validation data delineated in Appendix B – Initial Experimental Database for Fuel Performance and Radiological Release/Transport Methodology.

To have confidence in an EM's predictions, it must undergo rigorous review – a process called software Verification, Validation and Uncertainty Quantification. The first step is verification which ensures that the model (differential) equations are correctly solved, and the numerical solutions are consistent with the analytical solutions. The next step is validation where accuracy of the EM is evaluated by comparing the predictions with data obtained from relevant Separate Effects Tests and Integral Effects Tests. Finally, all analyses require an estimate of error and uncertainty in the prediction for an application. All the verification, validation, and uncertainty quantification activities are application dependent. The V&V of the EMs are included in its safety software assessment plan guided by the TerraPower's Acquired Software Quality Assurance Plan under Safety Analysis and Risk. Uncertainty quantification for the Natrium safety analyses is addressed in a Safety Uncertainty Quantification and Margin Assessment Methodology

Ongoing work in these areas is planned to be complete prior to TerraPower's submittal of an operating license application, and that information will be included in a future licensing submittal.

## 6.2 EM Conservatism Summary

In application of the methodologies discussed in this report for analysis of DBAs with potential fuel failure and radiological release, conservative assumptions about initial/boundary conditions, modeling parameters, and failure/acceptance criteria affect the outcome of calculations and predictions. Similar to the DBA without release calculations, conservative assessments of DBAs with fuel failure and release employ the following conservatisms in analysis of the in-vessel transient and partial flow blockage events [3]:

- Conservatisms in the form of direct biases are applied via input to nuclear data and model uncertainties, thermal-hydraulic models, and control system performance parameters for the representative events, and those which are applicable to the RAC performance. Selection of boundary conditions, isolation times, and other assumptions needs to ensure that the analysis is appropriately biased. The [[ ]](a)(4) steady-state analysis of heat-up events, for instance, uses the following set of biases in addition to the selected DBA biasing configuration:
    - [[
    ]](a)(4)

  - [[

    ]](a)(4)

- The plant initial and boundary conditions are conservatively selected within the operating band.
- The hot-pin PCT within the sub-assembly is conservatively assessed [[
  ]](a)(4)
- The TATNF screening criteria include conservatisms and margins that provide reasonable allowance that fuel pin failure will not occur when they are not violated.
- When the TATNF screening criteria are violated, the subsequent DSAW fuel performance analysis also contains conservatisms in evaluating the fuel failure.

The fuel misload analysis is adjusted for uncertainties in the final temperature distribution [[
]](a)(4) to determine the potential for fuel failures.

[[

]](a)(4)

An effort is underway to demonstrate that the conservative approach described above is sufficiently conservative for the Natrium design. Ongoing work is planned to be complete prior to TerraPower's submittal of an operating license application, and that information will be included in a future licensing submittal.

# 7 CONCLUSIONS AND LIMITATIONS

## 7.1 Conclusions

TerraPower is requesting NRC approval of the EM methodology plans documented in this report for use by future applicants utilizing the Natrium design as an appropriate and adequate means to evaluate DBAs with the potential for radiological release (as described in Section 2.4). This approval is subject to the limitations described below.

## 7.2 Limitations

All methodologies considered in this report share a set of similar limitations:

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 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, consideration of system interactions, and system conditions (which may affect the applicability of models or validation data).

# 8 REFERENCES

[1] NEI 18-04, Rev. 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Nuclear Energy Institute, 2019.

[2] NEI 21-07, Rev. 1, "Technology Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: For Applicants Utilizing NEI 18-04 Methodology," Nuclear Energy Institute, 2022.

[3] TP-LIC-RPT-0004 Rev. 0, "Design Basis Accident Methodology for In-Vessel Events without Radiological Release," TerraPower, 2023.

[4] TP-LIC-RPT-0003 Rev. 1, "Radiological Source Term Methodology Report," TerraPower, 2024.

[5] TP-LIC-RPT-0011 Rev. 0, "Core Design and Thermal Hydraulic Technical Report," TerraPower, 2024.

[6] NAT-2806 Rev. 0, "Natrium Topical Report: Fuel and Control Assembly Qualification," TerraPower, 2023.

[7] TP-LIC-RPT-0008 Rev. 0, "Partial Flow Blockage Methodology," TerraPower, 2024.

[8] TP-LIC-RPT-0005 Rev. 0, "Radiological Release Consequences Methodology Topical Report," TerraPower & GE-Hitachi Technology, 2023.

[9] SAND2011-4145, "Sodium Fast Reactor Gaps Analysis of Computer Codes and Models for Accident Analysis and Reactor Safety," SNL, 2011.

[10] TP-QA-PD-0001 Rev. 14A, "TerraPower QA Program Description," TerraPower, 2023.

[11] ASME NQA-1-2015, "Quality Assurance Requirements for Nuclear Facility Applications," The American Society of Mechanical Engineers (ASME), 2015.

[12] RG 1.28, "QUALITY ASSURANCE PROGRAM CRITERIA (DESIGN AND CONSTRUCTION)," US NRC, 2017.

[13] [[

[14]

]][(a)(4)]

[15] RG 1.82, Rev. 3, "WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT," US NRC, 2003.

[16] ORNL-TM-4324, "Effect of Partial Blockages in Simulated LMFBR Fuel Assemblies," ORNL, 1973.

# 9 APPENDICES

## 9.1 Appendix A – Additional Details of the DSAW Process

### 9.1.1 [[

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]](a)(4)

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]](a)(4)

## 9.2 Appendix B – Initial Experimental Database for Fuel Performance and Radiological Release/Transport Methodology

Experimental data listed below can be used to assess important closure models and integrated performance of the EMs developed to analyze fuel performance and in-vessel DBA events potentially involving fuel failure and/or radiological release. The lists are only preliminary and are retained here for historical purposes as they were used to inform the initial PIRT development and the subsequent experimental database development.

**Table B-1. List of Experimental Data Related to Radionuclide Migration during Pre-transient Phase**

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[[

]](a)(4)

**Table B-2. List of Experimental Data Related to Radionuclide Release during a Cladding Rupture**

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