UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

June 9, 2025

Dr. Mirela Gavrilas
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: NATRIUM TOPICAL REPORT, “RADIOLOGICAL SOURCE TERM METHODOLOGY REPORT,” (NAT-9392 REVISION 0)

Dear Dr. Gavrilas:

During the 725th meeting of the Advisory Committee on Reactor Safeguards, May 6 through 9, 2025, we completed our review of the TerraPower Natrium Topical Report, “Radiological Source Term Methodology Report,” Revision 0, and the associated draft safety evaluation (SE). Our TerraPower Subcommittee also reviewed this matter on March 19, 2025. During these meetings, discussions with the Nuclear Regulatory Commission (NRC) staff and TerraPower were beneficial, as were the referenced documents.

CONCLUSIONS AND RECOMMENDATION

1. Natrium would be the first sodium-cooled fast reactor (SFR) to implement a functional containment strategy. Although the Natrium functional containment shares many similarities with previous SFR containment designs, additional justification for departures from historical precedent is warranted during the upcoming construction permit application (CPA) review, because of the lack of detailed information and the use of non-safety designations for certain important components.

2. Sufficient data exists to support a mechanistic source term for more likely events inside the design basis (i.e., Anticipated Operational Occurrences (AOOs) and Design Basis Accidents (DBAs)). However, were eutectic fuel melting to occur (at ~700°C) in low-frequency events, the uncertainty in both the accident phenomenology and the release and transport of fission products in the vessel would significantly increase. These uncertainties make it difficult to assess the adequacy of the proposed source term methodology for fuel melt events. Addressing the uncertainties noted in this letter as part of the source term estimates in an operating license application will help establish sufficient confidence.

3. The SE should be issued, and the staff should consider the limitations noted in this letter during their CPA review.

BACKGROUND

The topical report outlines the methodology that TerraPower intends to use for calculating radiological source terms for the Natrium design. The applicant is adhering to the guidance of Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents of Nuclear Power Reactors,” Revision 0, Regulatory Position 2, to develop “attributes of an acceptable source term.” The methodology employs RG 1.203, “Transient and Accident Analysis Methods,” Revision 0, to guide the development and validation of their overall evaluation model. While the applicant does not plan to meet verbatim conformance with RG 1.203, they regard it as best industry practice.

Major Source Terms Considered in the Design

The design considers several major source terms, including:

- Releases from normal operation due to defective fuel, sodium activation, and tritium generation and release.
- Leaks from various plant systems, such as the cover gas cleanup system, the sodium cleanup system, the intermediate heat transport system, and the gaseous radioactive waste system.
- Source terms from licensing basis events, including anticipated operational occurrences, design basis events, and beyond-design-basis events, as well as other quantified events.
- Releases from fuel handling accidents in the vessel, in the ex-vessel storage tank, the spent fuel pool, the washing station, and during fuel transfer.

Functional Containment Strategy

TerraPower is implementing a functional containment strategy in their design, consisting of:

1. The Primary Functional Containment Boundary, defined as the minimum set of barriers encompassing the core and primary system that prevent a release of radionuclides from exceeding regulatory limits.
2. Enveloping Barriers, defined as structures, systems and components (SSCs), or portions of SSCs, that in the event of a leak or failure of a primary barrier, provide a backup radionuclide retention function to the Primary Functional Containment Boundary it envelopes.

For in-vessel events, the primary barriers are the reactor vessel and the reactor vessel head. The enveloping barriers include the physical structural enclosures surrounding the reactor vessel. For other sources of radionuclides in processing systems and the spent fuel pool, the barriers are different but usually involve an inert cell or enclosure.

Within the primary and enveloping containment barriers, TerraPower considers radionuclide removal mechanisms such as aerosol scrubbing via pool bubbles or aerosol deposition to be phenomena of the mechanistic source term analysis and not barriers of the functional containment. For example, the metallic fuel cladding, the sodium coolant, and the cover gas volume above the sodium coolant may provide significant attenuation affecting fission product release and transport from the primary containment boundary.

Evaluation Model Development

The Natrium evaluation model consists of computer codes to model the transport of radionuclides from the fuel to the environment. These codes use simplified aerosol transport models similar to those in the NRC's RADTRAD code for in-vessel and ex-vessel postulated events. A proprietary thermal hydraulics code is used to model transport within and from functional containment volumes. A severe accident code is planned to be used for any event in which sodium interacts with water (e.g., events in the spent fuel pool) or air (e.g., sodium release from the vessel in a large leak to the reactor building). Where underlying data on the release and transport behavior of fission products are missing or uncertain, the applicant plans to use conservative assumptions as part of the evaluation model development.

As a preliminary step in the development of the evaluation model, TerraPower generated Phenomenon Identification and Ranking Tables (PIRTs) for three specific events: drop of fuel in the spent fuel pool (a fuel handling accident), a sodium process system leak, and an unprotected loss of flow with degraded pump coastdown (an "other quantified event" lower in frequency than a beyond-design-basis event). Numerous highly ranked phenomena were identified in the process.

Fuel Drop Event. For the fuel drop event in the spent fuel pool, the effects of water interaction with the sodium bond in the fuel and the associated effects on fission product release were identified as important. The water can provide chemical energy and change the volatility of fission products released from the fuel. Thermomechanical analysis is planned to calculate fuel rod failure, and thermodynamic evaluations are planned to determine fission product chemical form. Experiments are planned to characterize the source term for this type of event.

Unprotected Loss of Flow. For the unprotected loss of flow event, the applicant identified significant uncertainty in the accident progression. The migration behavior of fission products from the fuel to the gap and upper plenum is known for some fission products like cesium but not for others like strontium. Upon cladding failure, the molten sodium bond at the top of the rod containing fission products, argon fill gas, and fission product and sodium vapors are released as a mixture of bubbles and aerosols. The bubbles and aerosols are transported through the sodium coolant where some attenuation occurs. Aerosols and vapors are subsequently transported into the cover gas where additional aerosol deposition and vapor condensation occur. Additional aerosol deposition is also anticipated in the structural enclosures above the reactor vessel as leakage from the cover gas of the reactor vessel occurs.

STAFF SAFETY EVALUATION

The staff SE focused on reviewing the evaluation methodology to determine its completeness relative to that recommended in RG 1.203. They provided many limitations and conditions related to the use of the methodology only for sodium-bonded metal fuel in a Licensing Modernization Project (LMP) based license application and limitations associated with the framework being neither fully developed nor validated at this point. The staff will review specific inputs and implementation of the source term evaluation model, including treatment of uncertainty in their review of future license applications.

DISCUSSION: FUNCTIONAL CONTAINMENT

The Natrium design is the first application of the functional containment concept to an SFR. Changes in the containment approach from previous SFR designs warrant careful attention to ensure that any deviations from precedent are justified. Despite its characterization as a "functional containment," the Natrium containment appears to be largely consistent with prior approaches. A representative SFR containment that shares many features with the Natrium functional containment barriers is described in Section 6.1.6 of Reference 4. When compared to the representative SFR containment approach, the Natrium functional containment:

a) Includes similar barriers, although Natrium designates some of the barriers as "enveloping" barriers and treats them as either non-safety related or non-safety related with special treatment. The Natrium topical report did not make clear what compromises to barrier integrity might be associated with the non-safety designations relative to prior designs that defined these barriers as safety related. We suggest the significance of the non-safety designations be clarified during the CPA review. The NRC staff should decide which of these compromises are acceptable and why.

b) Allows a higher leakage rate through some of the barriers. TerraPower specifically stated that allowable leakage through the head access area enclosure may be 10 vol% per day, as compared to 1 vol% per day in the SFR containment discussed in Reference 4. These allowable leakage rates are determined based on the mechanistic source term analyses, including consideration of very low frequency accident scenarios (termed "other quantified events" or (OQEs)) that would be screened from consideration as a licensing basis event by the LMP frequency-consequence curve. Details on OQE scenario selection, or the leakage acceptance criteria that might be applied for them are outside the scope of the topical report. We suggest that a more fulsome explanation for higher allowable leakage rates be provided during the CPA review.

DISCUSSION: PHYSICS AND CHEMISTRY OF FISSION PRODUCT RELEASE AND TRANSPORT IN SODIUM FAST REACTORS WITH METALLIC FUEL

Because the staff in its review focused on how the evaluation model framework developed by TerraPower complies with the guidance in RG 1.203, our assessment will instead focus on the underlying technical data (i.e., physics and chemistry that influence fission product release and transport) that forms the basis for input to the evaluation model. We followed this approach in large part because of the Commission approved direction in SECY-93-0092 that "sufficient data should exist to provide adequate confidence in the mechanistic approach," used to establish the mechanistic source term.

Fission Product Release from Metal Fuel

Overall, data on fission product release from metallic fuel at high burnup is very sparse. There is a lack of any systematic testing of metallic fuel as a function of temperature and burnup as has been done in other fuel systems. As a result, conservative estimates of release fractions of key fission products as a function of temperature have been established in Reference 5. The conservative release estimates suggest that the source term data can be used to support safety assessments under normal operation and for events that do not lead to fuel melting.

However, were the fuel to melt (i.e., above ~700°C), metallic fuel has been demonstrated not to be a strong barrier in terms of fission product retention. For low-frequency events where the fuel temperature might exceed 1100°C, recommended release fractions are 30% of the iodine, 100% of the cesium, 15% of the barium, 20% of the strontium, 10% of the cerium, and 30% of the lanthanide fission products. There is some evidence of retention of iodine and tellurium in metallic fuel, but the chemical form is not known with certainty. Additional post-irradiation examination of previously irradiated metal fuel pins could provide useful additional data to strengthen this database.

Fission Product Transport in Sodium

The transport of fission products through sodium depends on their physical form: gases, condensable vapors, and aerosols. Noble gases have low solubility in sodium and are not expected to condense. They are modeled to transit through the pool to the cover gas region. Any condensable fission product (e.g., cesium) and sodium vapors from the melt expulsion upon fuel failure are expected to condense in the sodium pool. Aerosols are carried by the gas (both noble gases and fuel pin fill gas) and transit through the approximately 6 meters of sodium above the top of the fuel to the cover gas region. During transit through the pool, gravitational settling, inertial deposition, and agglomeration are anticipated to reduce the aerosol source term entering the cover gas region.

Reference 8 states a major uncertainty is the amount of scrubbing that can be credited as bubbles transit through the sodium pool. For example, the need to account for "the potential to bypass the pool by transport in noble gas bubbles" was described in Section 3.1.3 of this report. We note that experiments (Reference 6) have been performed after Reference 8 was published to specifically evaluate the transport of medium and lower volatile fission product aerosols in gas bubbles through both water and sodium pools. Key variables include the pool depth, the particle size of the aerosol, and the gas velocity. Attenuation factors in the sodium pool were less than 10 over a range of aerosol sizes when the temperature of the pool and aerosol mixture was about 300°C.

It should be noted that the bubble transport model in References 7 and 8 has a decontamination factor for inertial deposition that is exponential with particle size. This dependence may greatly overpredict the ability of the sodium pool to scrub out aerosol particles if the model is used beyond the range of experimental data (approximately 3 microns). Within the bounds of the testing, these experiments provide a sound foundation for the source term model when fuel rod failure does not involve high temperature ejection of the melt/aerosol mixture from the fuel rod.

However, no experiments have been performed for cases where the melt/aerosol mixture was greater than 300°C, even though much higher temperatures are anticipated in the fuel rod and the coolant during an unprotected loss of flow event. The staff should consider if a limitation and condition recognizing these bounds in the experimental database is warranted.

Calculations of the bubble behavior when higher temperature melts are expelled from the fuel rod (as in low frequency unprotected loss of flow events) result in very large attenuation factors due to condensation in the sodium pool. However, data on bubble behavior does not exist under these high-temperature conditions. The physics involved in the expulsion of the melt from the fuel and mixing within the sodium coolant is complex, and some potentially important phenomena related to aerosol formation are not included in the evaluation model.[1] In our judgement, these phenomena need not be included in the evaluation model. Instead, sensitivity studies assuming all of the ejected melt exists as a very small aerosol (0.01 microns) versus a vapor, can be used to bound the behavior. This will result in a greater source term to the cover gas space, but the aerosol attenuation in that volume may offset this greater aerosol source.

Aerosol Deposition in Cover Gas, Enveloping Enclosures and Reactor Building

Once in the cover gas or upon leakage from the reactor vessel into the enveloping barriers, such as the head area access enclosure or the reactor building, aerosol transport phenomena like gravitational settling and agglomeration are important. Vapor condensation on cooler surfaces may also occur. Numerous experiments (References 9 and 10) of sodium oxide aerosol behavior in large volumes have been conducted in the past, and code-to-data benchmarks with CONTAIN-LMR have been used to establish a conservative deposition rate for use in simpler aerosol transport codes. In addition, TerraPower plans to use settling rates based only on the radioactive component of the aerosol (and not the non-radioactive component), which is conservative.

The thermal hydraulic behavior in a large volume like the reactor building or other structural enclosure can have a large influence on the transport behavior of fission products. Flows are expected to be driven by natural convection or chemical reactions (e.g., sodium-air interactions) instead of the strong pressure-driven flows in a light water reactor severe accident. Natural circulation and its associated uncertainty are important in establishing the effectiveness of such barriers.

Source Term Perspectives from Trial Calculations

Results of trial calculations with extensive sensitivity analyses in Reference 8 for a generic metal-fueled SFR provide important results about the physics and chemistry of the source term in SFRs, especially in the more severe lower frequency events. Bubble transport in the sodium coolant was found to have the highest impact on the overall source term, followed by fission product release from the fuel. With metal fuel, actinide release is significantly more important than in other reactor systems. Aerosol deposition and leakage from the reactor head into containment had medium importance. Pool vaporization and radioactive decay had low importance. While these results may not be fully applicable to Natrium, they can help focus future reviews and provide the staff with a better understanding of where the greatest uncertainties lie.

In our judgement, sufficient data exists to support a mechanistic source term for the more likely events within the design basis (e.g., AOOs and DBAs). However, were eutectic fuel melting to occur (at ~700°C) in low frequency events, the uncertainty in both the accident phenomenology and the release and transport of fission products in the vessel would increase significantly. These uncertainties make it difficult to assess the adequacy of the proposed source term

[1] At high temperatures, thermal radiation will quickly transfer heat from the ejected melt to the surrounding pool. The rapid drop in temperature will lead to supersaturation of the sodium and cesium vapors because traditional mass transfer will not be fast enough to reduce their partial pressure via condensation. This will promote the nucleation of fine aerosol particles instead of condensing the vapor on the surface of the bubble. This phenomenon occurs in many situations where high temperature metal vapors are cooled rapidly including aerosol formation from Ag-In-Cd control rods in Light Water Reactor (LWR) severe accidents and in liquid metal magnetohydrodynamic (MHD) applications.

methodology for fuel melt events. Addressing the impact of the uncertainties noted in this letter as part of source term estimates in a license application will help establish sufficient confidence. Our evaluation of the underlying data supporting radionuclide retention as modeled in the source term methodology indicates:

a. The fuel would not serve as a strong barrier to fission product release were it to melt during more severe low-frequency events.

b. The sodium coolant significantly retains fission products, but there are limitations in the underlying data at high temperatures that might be reached in some low-frequency events.

c. The anticipated aerosol behavior in the cover gas and enclosures surrounding the reactor vessel is well supported by the extensive sodium oxide aerosol experiments conducted in the past.

d. Sodium-water interactions in a spent fuel pool drop event can change the source term. Planned experimental work will help to establish a credible source term for this event.

SUMMARY

TerraPower plans to use a functional containment strategy as a basis for calculating mechanistic source terms for the Natrium reactor. This letter compares this containment approach to historical precedent in previous SFR designs. An assessment of the underlying physics and chemistry of fission product release from fuel and aerosol transport in the sodium pool, cover gas, and additional enclosures surrounding the reactor vessel is also provided highlighting uncertainties in the overall database.

The staff stated they were reviewing the details of the functional containment design and the source term during the CPA and subsequent licensing reviews. We look forward to further discussions on this topic. The SE should be issued, and the staff should take note of the limitations discussed in this letter.

We are not requesting a response to this letter.

Sincerely,

Walter L. Kirchner Signed by Kirchner, Walter on 06/09/25

Walter L. Kirchner
Chairman

Enclosures:
1. List of Acronyms
2. Additional Comments By ACRS Member Robert Martin

REFERENCES

1. TerraPower LLC, Topical Report, “Radiological Source Term Methodology Report,” Revision 0, August 11, 2023 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23223A235).

2. U.S. NRC, Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents of Nuclear Power Reactors,” Revision 0, July 31, 2000 (ADAMS Accession No. ML003716792).

3. U.S. NRC, Regulatory Guide 1.203, “Transient and Accident Analysis Methods,” Revision 0, September 30, 2005 (ADAMS Accession No. ML053500170).

4. Idaho National Laboratory, “Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors,” INL/EXT-14-31179, Revision 1, December 2014.

5. D. Grabaskas, et al., “Regulatory Technology Development Plan Sodium Fast Reactor: Mechanistic Source Term – Metal Fuel Radionuclide Release,” ANL-ART-38, February 2016.

6. Kyle F. Becker and Mark H. Anderson, “Experimental Validation of Simplified Radionuclide Transport Bubble Scrubbing Code in Sodium Coolant Pool,” Nuclear Engineering and Design, Volume 403, March 2023, 112137.

7. Gen Jiang, et al., “Development of a semiempirical model for the aerosol scrubbing in a sodium pool,” Progress in Nuclear Energy, 184, (2025), https://doi.org/10.1016/j.pnucene.2025.105699).

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9. IAEA, “Modelling and Simulation of the Source Term for a Sodium Cooled Fast Reactor Under Hypothetical Severe Accident Conditions,” IAEA TECDOC-2006, 2022.

10. D. Grabaskas, et al., “Review and Assessment of Available Data Regarding the Behavior of Sodium Aerosols,” ANL-NSE-22/68, September 2023.

List of Acronyms

AOO	Anticipated Operational Occurrence
CPA	Construction Permit Application
DBA	Design Basis Accident
EOP	Emergency Operating Procedures
LMP	Licensing Modernization Project
LWR	Light Water Reactor
NRC	Nuclear Regulatory Commission
MHD	Magnetohydrodynamic
MST	Mechanistic Source Term
NSRST	Non-Safety-Related with Special Treatment
NEI	Nuclear Energy Institute
OQE	Other Qualified Event
PIRT	Phenomenon Identification and Ranking Tables
PRA	Probabilistic Risk Assessment
RADTRAD	Radionuclide Transport and Removal Dose
RIPB	Risk-Informed Performance-Based
RG	Regulatory Guide
SE	Safety Evaluation
SFR	Sodium-Cooled Fast Reactor
SSCs	Structure, Systems and Components
TR	Topical Report

Additional Comments by ACRS Member Robert P. Martin

Consistent with statements in the Committee's letter addressing TerraPower's Mechanistic Source Term (MST) Topical Report (TR), I support the issuance of the staff's safety evaluation.

Expert Elicitation and Phenomena Identification and Ranking Table Transparency

The use of expert elicitation to develop Phenomena Identification and Ranking Tables (PIRTs) is a well-established practice in preparing evaluation models used to demonstrate safety-in-design. As reflected in NUREG-5249 and RG 1.203, PIRTs are not merely advisory; they form an integral part of model development, influencing the selection of dominant phenomena, experimental priorities, and validation strategies. TerraPower applied RG 1.203 to their MST evaluation, but did not identify the experts who participated in the associated PIRT.

When such models are submitted in support of licensing applications, whether for MST evaluations or other safety analyses, the credibility and defensibility of the expert elicitation process depends on transparency regarding the identity and qualifications of contributing experts. The NRC precedent, established via NUREG-1563, NUREG/CR-5074, NUREG-5249, and SRM-COMGEA-11-0001, consistently affirmed that such transparency is essential to ensure traceability, accountability, and legal robustness. I recommend that the NRC and applicants uphold this standard in all regulatory applications that rely on expert elicitation, including that associated with TerraPower's MST TR.

June 9, 2025

SUBJECT: NATRIUM TOPICAL REPORT, "RADIOLOGICAL SOURCE TERM METHODOLOGY REPORT" (NAT-9392 REVISION 0)

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