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

July 25, 2025

Mr. Michael F. King
Executive Director for Operations, Acting
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: X-ENERGY, LLC'S, XE-100 LICENSING TOPICAL REPORT MECHANISTIC SOURCE TERM APPROACH, REVISION 3

Dear Mr. King:

During the 727th meeting of the Advisory Committee on Reactor Safeguards, held from July 9 through 10, 2025, we completed our review of the X-energy, LLC, (X-energy) topical report on the "Mechanistic Source Term Approach," (000632, Revision 3) and the associated draft safety evaluation (SE). Our X-energy Subcommittee also reviewed this matter on June 3, 2025. During these meetings, discussions with the Nuclear Regulatory Commission (NRC) and X-energy staff were beneficial, as were the referenced documents.

CONCLUSIONS AND RECOMMENDATIONS

1. X-energy is developing a sequence-specific mechanistic source term (MST) through the use of a functional containment concept for their Xe-100 pebble bed reactor. This functional containment consists of the tristructural isotropic (TRISO) fuel kernel and coatings, the pebble matrix, and the helium pressure boundary.

2. The topical report describes a reasonable conceptual plan for the development of an MST methodology and the staff SE should be issued.

3. As the methodology is further developed and implemented in future licensing applications, there are several concerns and gaps that will need to be addressed that we identify in this letter.

4. Validation of the source term methodology will be challenging due to the numerous models involved, their complexity, gaps in the historical database, and residual uncertainties associated with the constitutive relations (material properties). Furthermore, the uranium dioxide (UO_2) TRISO fuel performance models do not adequately describe uranium oxycarbide (UCO) TRISO fuel performance.

5. A parallel semi-empirical approach, tied more directly to the statistically significant failure data from the U.S. Advanced Gas-cooled Reactor (AGR) UCO TRISO testing program with appropriate margins, may overcome these shortcomings.

BACKGROUND

Functional Containment Strategy

Consistent with prior high temperature gas reactor designs, such as the Modular High-Temperature Gas-Cooled Reactor (MHTGR), X-energy is implementing a functional containment strategy to retain fission products. The functional containment consists of the fuel kernel, the silicon carbide (SiC) and pyrocarbon (PyC) layers of the TRISO coating, the fuel matrix surrounding the particles in the pebble, the fuel free zone on the outside of the pebble, and the helium pressure boundary. No credit is taken for fission product retention in the reactor building.

The release of fission products is calculated based on the as-manufactured heavy metal contamination and SiC defects, in-service failures under irradiation, and incremental failures during licensing basis events (LBEs). For certain radionuclides, such as silver, diffusion through intact coatings is also considered. Additionally, deposition on the surfaces of the helium pressure boundary (also known as plateout) and dust in the system is accounted for, as well as resuspension or 'liftoff' of the deposited material during LBEs.

MST Model Development

The X-energy MST methodology, as implemented in the XSTERM computer code, comprises several models to describe the generation, release and transport of radionuclides from the fuel to the environment. The MST methodology includes:

- the thermal-hydraulic response of the fuel pebbles in the core under normal operation and LBEs,
- the production, decay, and transmutation of radionuclides in the pebble and their transport to the coolant, including the effects of dust,
- the transport to the reactor building, and
- evaluation of radiological dose to compare to regulatory limits.

The following are the key models in XSTERM:

- **Particle failure model** considers pressure vessel failure, kernel migration, fission product corrosion, thermal decomposition, inner pyrolitic carbon (IPyC) layer cracking and manufacturing defects.

- **Solid fission product release and transport** is based on diffusion through the kernel, coating layers and pebble matrix using detailed nodalization of each pebble in the analysis.

- **Thermodynamics model** calculates temperatures in all of the reactor components necessary to support an MST.

- **Steady state fission gas release** is calculated using two different (German and U.S.) release models that account for diffusion (a) through the kernel and buffer porosity for failed and defective fuel, and (b) through the matrix for the initial heavy metal contamination.

- **Dust production** is estimated using the measured particle size and estimated generation rates for the German Pebble Bed AVR scaled to the Xe-100.

- **Fission product transport, deposition and liftoff behavior in the helium pressure boundary** is based on models for sorption of fission products on dust, plateout of the dust and condensable fission products (e.g. iodine, cesium and strontium) during normal operation, and subsequent liftoff under postulated accidents.

- **Core corrosion models** describe the response of the core to oxidation events, including models for mass transport and chemical reactions during an air or steam ingress event, using data on air and steam oxidation behavior of graphite and data from industrial chemical synthesis technologies.

- **Tritium behavior model** tracks tritium production, decay, permeation, sorption on graphite, and release upon oxidant ingress into the core.

- **Reactor kinetics model** establishes the steady state condition and reactivity response to steam generator tube rupture events.

- **Atmospheric dose calculations** are based on traditional Gaussian plume dispersion (and appear to be consistent with NRC dose models for use in siting and control room habitability assessments).

STAFF SAFETY EVALUATION

The staff, in the SE, focused on physical phenomena of interest at a high level and the adequacy of the plan to develop the MST methodology, finding that the overall plan appears reasonable. However, they noted that the verification and validation plan is not based on a Phenomena Identification and Ranking Table. The staff did not examine details of the individual models of XSTERM, nor did they make any conclusions about the acceptability of the models, as they are still under development. The staff expects to perform a detailed review of XSTERM as part of a subsequent licensing application.

DISCUSSION

Verification and Validation Plans

X-energy intends to verify and validate the models in XSTERM using a combination of German and Chinese data related to UO_2 TRISO fuel, U.S. data related to UCO TRISO fuel, and analytical benchmark problems from HTR-10[1]. Separate effects data are utilized to validate individual models, such as the SANA[2] pebble bed heat transfer testing, German VAMPYR[3] plateout data, and Chinese Lifting Line Platform Facility data on dust generation. The following sections outline our assessment of these plans, their limitations, and provide recommendations to enhance the validation process and alert the staff and applicant to additional data not mentioned in the topical report.

[1] 10 MWt prototype high-temperature gas-cooled, pebble-bed reactor at Tsinghua University in China
[2] Composed of the German words for Secure Decay Heat Removal
[3] VAMPYR-I is a hot gas sampling tube installed in the German AVR reactor

Defects, Failure Fractions and Performance Envelopes

The source term from TRISO fuel in a high temperature gas-cooled reactor is a strong function of the initial level of manufacturing defects (heavy metal contamination, SiC defects), the in-service failures under irradiation, and the incremental failures during postulated accidents. Because of the importance to the overall functional containment strategy, the initial fuel failure fractions used in the X-energy safety analysis approach must bound actual manufacturing data. The values used for these failure fractions should align with the results from the AGR UCO TRISO testing, along with any additional fuel qualification testing planned by X-energy. Finally, the service conditions (burnup, temperature, and fast fluence) experienced during and postulated LBEs should remain inside the testing envelopes associated with the U.S. AGR UCO database, as supplemented by any additional testing planned on UCO by X-energy.

Fission Product Release Groupings

The database on fission product behavior in TRISO fuel is limited to measurements of noble gases, cesium, strontium, europium, silver and in some limited cases, iodine. Consequently, grouping of fission products into classes based on chemical or volatility considerations is essential for estimating total source terms for high temperature gas-cooled reactor (HTGR), similar to other reactor technologies. To align with previous HTGR safety and source term assessments (Reference 3), iodine and tellurium releases should be modeled as noble gases, while europium should be treated analogously to strontium. Lower volatile fission products, such as lanthanum and cerium, are not expected to be released from UCO TRISO fuel, according to measurements from the AGR program. However, the specific isotope list used in XSTERM is consistent with previous HTGR safety assessments.

Steady State Fission Gas Release

The topical report did not mention steady state fission gas release data collected over a large range of temperature and burnup from the U.S. UCO TRISO program (AGR-3/4) that contained failed fuel. These data have been published (Reference 4) and can be used either directly for fission gas release from exposed kernels or used to validate models for this part of the source term.

Fuel Performance

The fuel performance validation plan has three major shortcomings: (a) the models predominantly describe UO_2 rather than UCO TRISO fuel performance, (b) the failure mechanism observed in AGR UCO TRISO irradiations and heating tests[4] is not accounted for in the fuel failure model, and (c) the uncertainties in material properties required to describe fuel behavior are significant enough to make validation challenging. Additional technical details of these concerns are found in the Appendix to this letter.

Overall, considering the significant anticipated performance margin in the Xe-100 core and the concerns mentioned above, it may be more beneficial to directly use the measured statistically significant failure data from AGR-1 and AGR-2 (Reference 5) with appropriate margin as an estimate for the fuel failure fraction in source term calculations. Given the ongoing challenges

[4] The failure mechanism appears to be associated with buffer shrinkage inducing a crack that propagates through the inner PyC layer. The crack provides a pathway for palladium from the kernel to attack or corrode the SiC layer causing it to fail.

in validating detailed fuel performance and fission product transport models for UCO TRISO fuel, this kind of semi-empirical approach may offer a more practical and transparent path for source term estimation in the near future. Until uncertainties in mechanistic modeling are resolved, this approach may provide a more defensible basis for estimating Xe-100 source terms.

Fuel Behavior under Reactivity Events

Although fuel performance under reactivity events is not mentioned in the topical report, test data exist (Reference 6). This may not be a problem for the Xe-100 design because of the low excess reactivity in the core and their reactivity control strategy. However, some mention of the fuel behavior under these conditions is worthwhile from a completeness perspective.

Dust

The models for dust generation, transport, deposition (plateout) and resuspension (liftoff) are highly complex and challenging to determine if they are conservative. Sensitivity studies examining the timing of dust liftoff relative to fission product release during the LBEs, varying the dust generation rate and performing calculations with and without the dust, are recommended to help establish the overall role of dust on fission product transport in the Xe-100 during postulated events and provide more confidence in the predictions.

Beyond the VAMPYR plateout testing in Germany, the report does not mention the extensive testing done in the COMEDIE[5] facility in the 1990s (Reference 7) to examine deposition and subsequent liftoff under various break sizes. Additionally, the applicant did not cite the large amount of data on resuspension of metallic aerosols (dust) in the aerosol literature (References 8, 9, 10) that could be useful for validation. Ultimately, validation through measurements of gaseous and metallic fission products during operation will be required to ensure specified acceptable radionuclide release design limits are being met.

Core Corrosion/Oxidation

The existing database on the response of TRISO fuels to air or steam ingress is quite limited. Some in-pile testing has been performed for short duration and at specific temperatures focusing primarily on fission gas measurements rather than the fission metals that tend to dominate the radiological dose. There are some limited data on loose particle testing in air that show high failure rates (Reference 6), albeit under conditions that are not representative of the Xe-100. However, there are no data on the effects of steam ingress on UCO TRISO coated particle fuel as would be expected in a steam generator tube rupture. The assumption in XSTERM that steam will not result in particle failure is unsubstantiated. During discussions in our subcommittee meeting, the applicant mentioned that in their simulations, the steam never reaches the particles due to the fuel-free zone on the surface of the pebble. This behavior will need to be validated.

Given the importance of these events, as highlighted by the results of the modular HTGR probabilistic risk assessment, the U.S. AGR TRISO program plans on testing fuel compacts in steam and air under a range of temperatures and partial pressures of oxidant (Reference 11). The testing will measure fission gas and metallic fission product release during such exposures. Historically, such testing of this nature had been planned in U.S. TRISO fuel

[5] High pressure COMEDIE test loop experiments

qualification programs for decades. However, it was never carried out due to the requirement for special furnaces and fission product detection systems. These systems have only recently been developed under the AGR program. The staff should remain cognizant of this testing program.

SUMMARY

X-energy is developing a sequence-specific MST through the use of a functional containment concept for their Xe-100 pebble bed reactor. This functional containment consists of the TRISO fuel kernel and coatings, the pebble matrix, and the helium pressure boundary. The topical report describes a reasonable conceptual plan for the development of an MST methodology and the staff SE should be issued. As the methodology is further developed and implemented in future licensing applications, there are several concerns and gaps that will need to be addressed that we identify in this letter.

We are not requesting a response to this letter.

Sincerely,

Walter L Kirchner Signed by Kirchner, Walter on 07/25/25

Walter L. Kirchner
Chairman

Enclosures:

1. Appendix: Technical concerns with Validation of UCO TRISO Fuel Performance Models as part of XSTERM

2. List of Acronyms

REFERENCES

1. X-Energy, LLC, "Xe-100 Licensing Topical Report Mechanistic Source Term Approach," Document ID 000632, Revision 3, March 4, 2025 (ADAMS Accession No. ML24131A146).

2. U.S. Nuclear Regulatory Commission, Letter to Kalinousky, X-Energy, LLC from Philpott, USNRC, "U.S. Nuclear Regulatory Commission Draft Safety Evaluation for X-Energy LLC's Xe-100 Licensing Topical Report Mechanistic Source Term Approach," Revision 3, May 6, 2025 (ADAMS Accession No. ML25064A595).

3. D. A. Petti, R. R. Hobbins, P. Lowry, and H. Gougar, "Representative Source Terms and the Influence of Reactor Attributes on Functional Containment in Modular High-Temperature Gas-Cooled Reactors," *Nuclear Technology*, Vol. 184, Nov. 2013.

4. Binh T. Pham, Jeffrey J. Einerson, Dawn M. Scates, John T. Maki, and David A. Petti, "AGR-1, AGR-2 and AGR-3/4 Release-to-Birth Ratio Data Analysis," INL/EXT-14-32970, Revision 2, June 2019.

5. EPRI, "Uranium Oxycarbide (UCO) Tri-structural Isotropic (TRISO) Coated Particle Fuel Performance Topical Report," EPRI-AR-1(NP), 2019.

6. IAEA, "High Temperature Gas Cooled Reactor Fuels and Materials," IAEA-TECDOC-CD-1645, March 2010.

7. D. L. Hanson and A.A. Shenoy, "In-Pile Loop Tests to Validate Fission Product Transport Codes," Proceedings HTR2006: 3rd International Topical Meeting on High Temperature Reactor Technology, October 1-4, 2006, Johannesburg, South Africa.

8. B. J. Merrill and P. W. Humrickhouse, "Aerosol Resuspension Model for MELCOR for Fusion and Very High Temperature Reactor Applications," INL/EXT-10-19683, January 2011.

9. Bruno Gonfiotti and Sandro Paci, "Implementation and validation of a resuspension model in MELCOR 1.8.6 for fusion applications," *Fusion Engineering and Design* Volume 122, November 2017, Pages 64-85.

10. Liwen He, Lili Tong, Xuewu Cao, "Investigation of aerosol resuspension model based on random contact with rough surface *Nuclear Engineering and Technology*," Volume 55, Issue 3, March 2023, Pages 989-998.

11. J.Stempien, "AGR Irradiated Specimen Air/Moisture Heating Test Plan," INL/MIS-17-43986, PLN-5934, December 2019.

12. Blaise P. Collin, David A. Petti, Paul A. Demkowicz, and John T. Maki, "Comparison of fission product release predictions using PARFUME with results from the AGR-1 safety tests," *Nuclear Engineering and Design,* Vol. 301 (2016) pp. 378–390.

13. Blaise Collin, David A. Petti, Paul A. Demkowicz, John T. Maki, "Comparison of silver, cesium, and strontium release predictions using PARFUME with results from the AGR-1 irradiation experiment," *Journal of Nuclear Materials*, 466 (2015) 426-442.

14. Lance L Snead, Takashi Nozawa, Yutai Katoh, Thak-Sang Byun, Sosuke Kondo, David A. Petti, "Handbook of SiC properties for fuel performance modeling," Journal of Nuclear Materials, Volume 371, No. 1, Pages 329-377, September 2007.

15. IAEA, "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," IAEA TECDOC 978, Nov. 1997.

16. William F. Skerjanc and Blaise P. Collin, "Assessment of Material Properties for TRISO Fuel Particles used in PARFUME," INL/EXT-18-44631, Revision 0, August 2018.

Appendix: Technical concerns with Validation of UCO TRISO Fuel Performance Models as part of XSTERM

The fuel performance validation plan in XSTERM has three major shortcomings: (a) the models predominantly describe UO_2 rather than UCO TRISO fuel performance, (b) the failure mechanism observed in AGR UCO TRISO irradiations and heating tests is not accounted for in the fuel failure model, and (c) the uncertainties in material properties required to describe fuel behavior are significant enough to make validation challenging.

UO_2 vs. UCO TRISO Fuel Performance. X-energy plans to validate the fuel performance models using a combination of irradiation and heating (safety) data from UO_2 (German irradiations HFR K5 and K6 in the Petten high flux research reactor) and UCO TRISO fuel (AGR-1 and AGR-2). While some of this data is applicable, the UO_2 behavior does not fully represent modern UCO TRISO fuel.

Specifically, many if not all the failure mechanisms modeled in XSTERM were historically observed in UO_2 TRISO fuel but have been deliberately engineered out of UCO TRISO fuel. This fuel development approach was adopted in the U.S. decades ago by modifying particle design, altering fabrication conditions, and limiting reactor service conditions. If these mechanisms are not relevant to UCO TRISO, it raises doubts about the effectiveness and completeness of using such a validation effort that relies on them.

Failure mechanism in AGR testing. The applicant plans to use the results of AGR-1 and AGR-2 in their validation efforts. However, it is important to note that the failure mechanism noted for UCO TRISO in AGR-1 and AGR-2[6] (Reference 5) is not modeled in any particle fuel performance code due to uncertainties in the material properties necessary to support the model. This problem is sufficiently difficult that advanced multiscale modeling efforts are underway in the Department of Energy's Nuclear Energy Advanced Modeling and Simulation (NEAMS) program as a "challenge problem."

Material Property Uncertainties. From a fission product release perspective, attempts to validate diffusional release models using data from AGR-1 and AGR-2 have had limited success. This is due to the low level of releases that were measured, the potential multiple sources of the fission products (heavy metal contamination, exposed kernels or through intact particles), and the uncertainties in the underlying diffusion coefficient database. (References 12 and 13)

The report also does not address many of the uncertainties in the thermomechanical material properties that significantly influence model predictions. For example, the SiC strength data in the topical report is based on German TRISO fuel and shows considerably greater strength than more recent UCO TRISO fuel (Reference 14). This difference is likely due to differences in the microstructure of the SiC layer. Additionally, the PyC creep and shrinkage rate data used in XSTERM that determine the survivability of the PyC layers are simple fluence-based estimates from old German testing. They do not represent the more contemporary data used in a recent International Atomic Energy Agency code benchmark (Reference 15) that are temperature and fluence dependent. These data are based on a compendium of historical U.S. and international data and are considered to be more representative of UCO TRISO fuel behavior. The uncertainties in these fundamental material properties are in part driven by the small scale of

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[6] The failure mechanism appears to be associated with buffer shrinkage inducing a crack that propagates through the inner PyC layer. The crack provides a pathway for palladium from the kernel to attack or corrode the SiC layer causing it to fail.

the samples and limitations in testing. These thermomechanical material properties (SiC and PyC strength data and PyC shrinkage/creep data) drive failure probability predictions (Ref 16), and the uncertainties are large enough to make validation of failure probabilities challenging.

LIST OF ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
AGR	Advanced Gas-Cooled Reactor
HTGR	High-Temperature Gas-Cooled Reactor
IPyC	Inner Pyrolitic Carbon
LBEs	Licensing Basis Events
MHTGR	Modular High-Temperature Gas-Cooled Reactor
MST	Mechanistic Source Term
NRC	Nuclear Regulatory Commission
PyC	Pyrocarbon
SE	Safety Evaluation
SiC	Silicon Caride
TRISO	Tristructural Isotropic
UO_2	Uranium Dioxide
UCO	Uranium Oxycarbide

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