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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 – 0001

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Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20444-001

SUBJECT: SAFETY EVALUATION OF THE KAIROS TOPICAL REPORT, KP-TR-012-P, REVISION 1, "KP-FHR MECHANISTIC SOURCE TERM METHODOLOGY"

Dear Mr. Dorman:

During the 691st meeting of the Advisory Committee on Reactor Safeguards, November 30 through December 2, 2021, we completed our review of the NRC staff's Safety Evaluation (SE) report on the Kairos Topical Report KP-TR-012-P, Revision 1, "KP-FHR Mechanistic Source Term Methodology." Our Kairos Subcommittee reviewed this matter on November 19, 2021. During these meetings we had the benefit of discussions with the staff and members of Kairos Power, LLC (Kairos). We also had the benefit of the referenced documents.

Conclusions and Recommendations

1. The topical report presents the methodology used by Kairos to calculate the mechanistic source term of fission products, activation products, and corrosion products produced in the Kairos Power fluoride salt-cooled high temperature reactor (KP-FHR) core. The approach is consistent with existing high-level regulatory guidance on source terms for advanced reactors.

2. Staff review of an application that employs this methodology will need to ensure that the assumptions on the number of failed pebbles as well as the experimental limitations related to tritium behavior in the molten salt coolant for the KP-FHR (Flibe[1]) and diffusion and trapping effects in graphitic components are adequately considered in conservative safety analyses and relevant sensitivity studies.

3. The staff SE does not require experimental validation of vaporization of fission products from Flibe. This has an important effect on the overall source term, and experimental validation data are needed to confirm the approach used by the applicant. The SE should be changed to address this concern.

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[1] Flibe is a mixture of lithium fluoride (LiF) and beryllium fluoride (BeF2), with a nominal chemical composition of 2LiF:BeF2.

Background

The topical report presents the methodology proposed by Kairos to calculate a mechanistic source term for fission products, activation products and corrosion products generated in the KP-FHR core. Mechanistic source term models are designed to calculate the transport and retention processes based on fundamental chemistry, thermodynamics, and kinetics of the reactor system and its environs. This topical report is to be used to calculate source terms during anticipated operational occurrences, design basis events, and design basis accidents. It does not cover beyond design basis events.

There is no NRC guidance on source terms specifically applicable to non-light water reactor (LWR) designs. However, generic guidance exists in current NRC documents including NUREG-0800, RG 1.183 (Regulatory Position 2), RG 1.145 and RG 1.194. SECY-16-0012 and SECY-93-092 provide additional information for calculating a mechanistic source term and its relationship to functional containment. Finally, RGs 1.232 and 1.233 provide additional considerations related to mechanistic source terms as part of establishing the licensing basis for advanced reactors. The approach in the topical report incorporates many of the key features of the high-level guidance found in these documents.

Discussion

The methodology relies on the following three inherent design features of the KP-FHR that are not intended to change as the design evolves: (a) the pebble handling system that is connected to the reactor head, (b) passive heat removal from the reactor vessel and natural circulation in the reactor vessel to remove decay heat, and (c) functional containment of radionuclides consisting of the TRISO-coated particle kernel and coatings, the molten salt coolant (Flibe), and confinement enclosures around the reactor. Given the strong radionuclide retention features of both TRISO fuel and Flibe, the fission product source term is anticipated to be much smaller than existing LWRs.

The topical report defines a number of sources of materials at risk (MAR) in the reactor design including radionuclides in the fuel and graphite pebbles, the molten salt coolant, and the graphite reflector as well as in the cover gas that will be located at the top of the reactor vessel. The methodology assesses release of radionuclides from each of these sources. A series of codes will be used as part of the methodology: SERPENT2 for fission product inventory, KP-BISON for fission product release from the fuel, KP-SAM for Flibe coolant transport, RADTRAD for transport in the gaseous spaces, and ARCON96 for atmospheric dispersion of the release and conversion to radiation doses.

Nine different Phenomena Identification and Ranking Tables (PIRTs) were used to evaluate the radionuclide release and transport phenomena that needed to be included in the methodology. The PIRTs covered fuel-related, coolant-related, graphite structural, and cover gas related phenomena for accidents involving intact and compromised coolant boundaries. The PIRTs also cover accidents involved in radioactive waste handling and storage systems and accidents in the pebble handling and storage system. Phenomena ranked “high importance” were evaluated across a range of knowledge levels. Many similar phenomena were identified among the different PIRTs. Many of the high importance and low knowledge phenomena were deemed to be precluded by design and thereby considered beyond design basis.

Limitation 8 in the topical report points to design features that limit the impact of various phenomena. Examples include significant air ingress into the reactor, Flibe/concrete interactions, and reaction of the coolant with a chemical reactant that could change the redox potential outside of allowable levels. This could suggest a potential for cliff edge effects once the design evolves and a broad range of accident scenarios are assessed.

The following sections describe key parts of the mechanistic source term with some clarifications and a change regarding selected limitations and conditions identified in the staff SE.

Fission Product Release from Fuel Pebble

The methodology addresses fission product release from the fuel pebbles based on conservative levels of fuel manufacturing defects and in-service failures, and diffusion through intact particles. These release mechanisms establish a circulating activity in the Flibe coolant. No credit is given for the retention capability of the matrix of the fuel pebble. The fission products are classified into groups based on similarities in their chemical behavior and assigned a representative fission product for the methodology assessment. Fission product release has been measured and associated diffusion coefficients derived for the following fission products: Ag-110m, Sr-90, Cs-134 and Cs-137, Kr isotopes and Xe isotopes, and I-131. It is important to note that no releases have been measured on some of the fission product groups (e.g., actinides, lanthanides, post-transition metals) from TRISO fuel. Because there are no data on the release and transport behavior for many of the chemical groups, the methodology conservatively assigns one of these measured fission products to represent these fission product chemical groups. This is expected to be conservative in most cases especially for the low volatile fission products. The diffusion coefficients for the measured fission products for Uranium Dioxide (UO_2) TRISO, as described in International Atomic Energy Agency (IAEA) high-temperature gas-cooled reactor (HTGR) documents, are used in the methodology. These coefficients when applied to Uranium Oxycarbide (UCO) TRISO have been shown to be overly conservative when estimating releases from UCO TRISO in HTGR applications. Note that there are no diffusion coefficients for Eu in the IAEA database. Such information is anticipated to be developed as part of the advanced gas reactor fuel development program.

In some accident scenarios, the methodology also considered fission product releases from a mechanically damaged pebble using airborne release fractions established in NUREG-CR-6410. Kairos suggests that the likelihood of TRISO fuel failure in design basis events, from other than manufacturing defects, could be negligible due to the robust TRISO fuel design. The evaluation of design basis events is not provided in this report and will be provided as part of safety analysis reports in the licensing for KP-FHR. Applicants using this report will need to justify the assumption on the number of pebbles being damaged. Numerous pebbles were identified as broken during the defueling of the AVR pebble bed reactor in Germany.

Tritium Behavior

Tritium is generated in the KP-FHR through activation of lithium in the molten salt and to a much less extent lithium impurity in the graphite and ternary fission in the fuel particle. The methodology considers the generation and transport of tritium in the salt and its sorption onto the core graphitic material (pebbles and reflector graphite) based on the solubilities and diffusivities in the salt and in the core graphitic materials.

Establishing diffusivities of chemical elements in liquids via experiment is very difficult given the impact that small thermal gradients can have on mixing a stagnant liquid. While measurements of diffusivities of tritium in Flibe have been performed, they may be highly uncertain.

Correlations of diffusivities and viscosities exist (e.g., Stokes Einstein relationship) and should be used to assess measured diffusivities in Flibe. Obtaining accurate estimates of tritium solubility in Flibe is also difficult given the general low solubility and the experimental bias that can be introduced when injecting tritium in Flibe to replicate in-situ tritium generation. Applicants applying this methodology should consider these experimental limitations when attempting to bound uncertainty ranges to be used in any conservative source term and/or sensitivity analysis.

Diffusion and trapping in the graphitic materials are also considered. Traps are physical or chemical aspects of the material that retard the transport of species in the material. They can be formed during fabrication (e.g., porosity, dangling bonds, or microstructural features) or under irradiation (e.g., vacancies or dislocations). Some traps can also be annealed out at high temperature. Traps are generally characterized by their concentration and their energy depth. These trapping parameters are based on research conducted by the fusion community. The methodology also considers release of the tritium from the structures in the event of a high temperature transient.

It is not clear that the trapping parameters are applicable to the graphitic material in the KP-FHR. The specific surface area and porosity vary significantly across the different grades of graphite. Graphite materials studied by the fusion community have not seen the high temperature heat treatment typical of "nuclear" graphite. The matrix of the fuel pebble is also not strictly graphite because it has not been graphitized at high temperature; it is considered to be more of a carbonaceous material.

While the energy depth of the traps in carbon-based materials are probably similar, the different heat treatments for the carbon-based materials in the KP-FHR and the associated radiation damage will affect the trap concentrations available for tritium to be sequestered. This will affect the inventory of tritium in these components of the KP-FHR core relative to that in the salt. In addition, there could be potential competition for the traps in the fuel pebbles with fission products released from the defective fuel. Applicants using this methodology will need to determine the concentration of fission products in the matrix relative to the trap concentration. At a minimum, sensitivity calculations will need to be performed to evaluate the impact of trap concentration on tritium retention in the system. Validation of these diffusion and trapping models will be required.

Tritium is anticipated to permeate into the secondary nitrate salt coolant system. A dehumidification system is proposed to remove tritiated water vapor from the nitrate salt and prevent its release to the environment. This approach could result in the generation of a low concentration tritium waste stream that could create a disposal challenge.

Radionuclide release and retention in Flibe

The methodology uses chemical thermodynamic principles to establish the release and retention of radionuclides in the Flibe coolant, including fission products released from the fuel, fission of tramp uranium and thorium in the salt, and corrosion products dissolved from surfaces in contact with the salt during operation (e.g., chromium). Some radionuclides will be soluble and remain as fluorides in the salt; others will remain as insoluble oxide; and still others are more noble, remaining as a metallic phase in the salt. The split between noble metals and fluorides depends on the redox potential. The metallic phases could precipitate in the salt or plateout on the cool parts of the system (in the intermediate heat exchanger). Gaseous and some halogen fission products and tritium are expected to have very low solubility in the salt and reside in the cover gas volume and associated gas cleanup system. The assumptions on tramp fissile material in the salt (from uranium contamination) and the potential for dispersed uranium in the fuel pebble to migrate into the salt may be overly conservative.

Solubility limits need to be measured for some key radionuclides to understand the margin to precipitation in the Flibe coolant. The solubility of only a few materials have been measured in the salt (e.g., BeO in Flibe). Of particular importance may be competition between fission products given the overall solubility limit in the salt.

The retentiveness of Flibe as a barrier for most fission products is as good as if not better than that for TRISO fuel. Traditional vaporization models are used to model volatilization from the salt as a function of temperature, relevant vapor pressure (fluoride, oxide or elemental) and concentration in the salt assuming ideality. When data on a particular chemical form of a radionuclide are unavailable or for simplicity or conservationism, data from the most volatile chemical form in that family is used to conservatively bound the vaporization. Although the approach is based on classical chemical fundamentals, as with the other components of the source term, experimental validation of vaporization rates by comparison with experimental data should be required for this important piece of the overall source term. The SE should be changed to address this concern.

Vaporization of beryllium fluoride and its condensation on cold spots in the cover gas system could impact worker safety. Experience at the Molten-Salt Reactor Experiment suggests that the condensed material is very friable and could be an aerosol source. While not radioactive, given the toxicity of beryllium, it could also be important for public safety assessments since it is anticipated to be the most volatile component of Flibe.

The methodology also considers the generation of aerosols during a pipe break or spill of coolant. Models for both aerosols generated from jet breakup and aerosols generated from splashing have been developed based on work obtained from the literature. The methodology also considers the release of graphite dust from erosion of the surface of pebbles by interaction with other pebbles and surfaces in the reactor core and in the pebble handling system during transit. Fission products in the dust particles is assumed to be at the same concentration as that anticipated in the pebble fuel matrix.

Radionuclide Transport in the Gas Space and Atmospheric Transport

The methodology considers all condensable radionuclides to be aerosol. Only gravitational settling and radioactive decay are considered using the RADTRAD code. The settling rate is based on classic aerosol physics and is based on a correlation derived from the ABCOVE test AB5 sodium aerosol experiments.

Atmospheric Dispersion

The methodology describes the calculations to be performed for atmospheric dispersion using the ARCON96 model and relevant X/Q values. The staff found the approach to be reasonable but noted that specific parameters in the calculation of accident radiological consequences (e.g., breathing rates, dose coefficients) will require staff review when the model is applied to a specific design.

Staff Assessment and Limitations of the Methodology

The staff found the overall methodology to be acceptable. They found the use of the concept of MAR and release fractions from each barrier to be reasonable. The approach to screening out *de minimus* inventories was acceptable to the staff. The computer codes to be used in the methodology are acceptable (subject to validation) and chemical modeling based on fundamental chemical principals was reasonable. The approach to aerosol generation and transport was also acceptable.

The topical report provides 8 self-identified limitations of the methodology related to:

1. Approval of the KP-BISON code,

2. Justification of the thermodynamic analysis and associated vapor pressure data,

3. Validation of the tritium transport model,

4. Confirmation of minimal ingress of Flibe into a fuel pebble,

5. Establishment of operational limits on circulating activity, concentration of radionuclides relative to solubility limits in the coolant, cover gas and radioactive waste systems,

6. Quantification of the transport of tritium in the secondary nitrate coolant salt,

7. Restriction to molten not solid Flibe, and

8. Restriction to the design features of the KP-FHR.

The staff agrees with these limitations and proposed two additional ones:

1. The use of the topical report is limited to the KP-FHR and is not applicable to other molten salt reactor designs given that the use of TRISO fuel in the KP-FHR design limits the potential for fission product release.

2. Additional information is needed to justify that the calculation of tritium absorption into graphite is not sensitive to assumptions on diffusivity and solubility in the Flibe.

Summary

The topical report presents the methodology used by Kairos to mechanistically calculate the source term of fission products, activation products and corrosion products produced in the KP-FHR core. The approach is consistent with existing high-level regulatory guidance on source terms for advanced reactors.

Staff review of an application that employs this methodology will need to ensure that the assumptions on the number of failed pebbles as well as the experimental limitations related to tritium behavior in Flibe and diffusion and trapping effects in graphitic components are adequately considered in conservative safety analyses and relevant sensitivity studies.

The staff SE does not require experimental validation of vaporization of fission products from Flibe. This has an important effect on the overall source term and experimental validation data are needed to confirm the approach used by the applicant. The SE should be changed to address this concern.

Sincerely,

Signed by Sunseri, Matthew on 12/20/21

Matthew W. Sunseri
Chairman

References

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