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

June 17, 2020

The Honorable Kristine L. Svinicki
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: SUMMARY REPORT – 673rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, MAY 6-8, 2020

Dear Chairman Svinicki:

During its 673rd meeting, May 6-8, 2020, which was conducted virtually due to the COVID-19 pandemic and the mandatory telework order, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following correspondence:

LETTERS

Letters to Margaret M. Doane, Executive Director for Operations (EDO), NRC, from Matthew W. Sunseri, Chairman, ACRS:

- Safety Evaluation for Topical Report KP-TR-006-P, Revision 1, "Scaling Methodology for the Kairos Power Testing Program," dated May 27, 2020, Agencywide Documents Access and Management System (ADAMS) Accession No. ML20142A301
- Safety Evaluation of the Kairos Topical Report KP-TR-005-P, Revision 1, "Reactor Coolant for the Kairos Power Fluoride Salt-cooled High Temperature Reactor," dated June 1, 2020, ADAMS Accession No. ML20148M230
- NuScale Areas of Focus – Probabilistic Risk Assessment and Emergency Core Cooling System Valve Performance, dated June 1, 2020, ADAMS Accession No. ML20149K596

MEMORANDA

Memoranda to Margaret M. Doane, EDO, NRC, from Scott W. Moore, Executive Director, ACRS:

- Documentation of Receipt of Applicable Official NRC Notices to the Advisory Committee on Reactor Safeguards for May 2020, dated May 18, 2020, ADAMS Accession No. ML20135H268
- Regulatory Guide, dated May 18, 2020, ADAMS Accession No. ML20135H269

- Proposed Rule: 10 CFR PART 71, "Harmonization of Transportation Safety Requirements with IAEA Standards," dated May 18, 2020, ADAMS Accession No. ML20135H271

HIGHLIGHTS OF KEY ISSUES

1. Safety Evaluation for Topical Report KP-TR-006-P, Revision 1, "Scaling Methodology for the Kairos Power Testing Program"

The Kairos topical report (TR) summarizes the scaling methodology used in their testing program to design scaled experiments that predict behavior in the prototypical Kairos fluoride-salt-cooled high-temperature reactor. This methodology will be used to perform scaling analyses as part of the evaluation model development and assessment process described in Regulatory Guide 1.203, "Transient and Accident Analysis Methods."

The TR provides a basis for obtaining experimental data using surrogate fluids instead of a lithium fluoride (LiF), beryllium fluoride (BeF_2) salt, FLiBe ($2LiF:BeF_2$). The high operating temperatures, power requirements, and toxicity hazards of working with FLiBe make the use of surrogate fluids beneficial for testing. The use of surrogate fluids enables direct and comprehensive measurements of the phenomena under investigation due to the higher compatibility of available, more accurate instrumentation (e.g., temperature, flow velocity, and pressure) in surrogate fluids versus prototypical molten salts at high temperatures. Kairos intends to use heat transfer oil and water as surrogate fluids for FLiBe in specific thermal-fluids tests. This report demonstrates that these surrogate fluids provide acceptable substitutes for FLiBe for certain types of scaled integral effects tests and separate effects tests, and that the important thermal-fluids characteristics can be properly scaled. This enables Kairos to perform scaled experiments with these surrogate fluids before final testing with FLiBe.

The staff evaluated the scaling methodology approach for:

- integral effects tests,
- separate effects tests, and
- use of surrogate fluids in scaled tests

For integral effects tests, Kairos addresses the scenarios of: forced circulation under steady-state normal operation, natural circulation transient evolution, and quasi-steady natural circulation. The overall approach includes: performing a top-down scaling by performing a control volume analysis on the reactor using conservation of energy and momentum equations; afterwards, conducting a bottom-up scaling to focus on and capture all important phenomena and associated processes within individual modules and components. The staff review has found this approach to be acceptable because it is consistent with the well-established H2TS scaling methodology of NUREG/CR-5809, and it uses non-dimensional equations to develop similarity/scaling parameters using the methodology of NUREG/CR-3267, "Similarity Analysis and Scaling Criteria for LWR's Under Single-Phase and Two-Phase Natural Circulation." For separate effects tests, Kairos addresses the treatment of: forced circulation fluid dynamics, convective heat transfer, conjugate heat transfer with solid structures, channel flow experiments, and the scaling of twisted-elliptical-tube heat exchangers. The staff finds this treatment acceptable because it includes the use of scaling parameters and values obtained from non-dimensional transport equations that are well established for single-phase fluid flow and heat transfer.

Kairos also addresses the use of surrogate fluids in Section 5 of the report. The staff finds this approach acceptable because the use of surrogate fluids is known to result in small scaling distortions in single-phase fluid flow and heat transfer.

Committee Action

The Committee issued a letter on May 27, 2020, with the following conclusion and recommendation:

a) Topical report KP-TR-006-P, with the limitations and conditions imposed by the staff SE report, provides an acceptable methodology to scale momentum and heat transfer phenomena for the Kairos reactor under normal operations and transient conditions.

b) The safety evaluation (SE) report should be issued.

2. Safety Evaluation of the Kairos Topical Report KP-TR-005-P, Revision 1, "Reactor Coolant for the Kairos Power Fluoride Salt-cooled High Temperature Reactor"

Kairos submitted this KP-TR-005-P and requested the NRC staff review and approve the FLiBe thermophysical properties provided in Table 1, "Thermophysical Properties of the KP-FHR Primary Coolant," and the reactor coolant specification provided in Table 4, "Design Specification for the KP-FHR Reactor Coolant," of the KP-TR-005-P. Final versions of these properties and specifications can be used by applicants in future licensing submittals under Title 10 of the *Code of Federal Regulations* (10 CFR) Parts 50 or 52.

KP-TR-005-P discusses the known properties of the FLiBe coolant and the constraints or limits under which the coolant must operate to (a) assure the coolant maintains its proper thermophysical properties, (b) control corrosion, and (c) limit reactivity effects. In this topical report, Kairos provides documentation to support that the information in Tables 1 and 4 can be used by an applicant to demonstrate compliance with Kairos Power Fluoride High Temperature Reactor (KP-FHR) Principal Design Criteria, subject to the Limitations and Conditions identified in the staff SE report. The information provided in Tables 1 and 4 of KP-TR-005-P establishes initial values for certain characteristics of the reactor coolant that will support unique design features of the KP-FHR as well as its safety and operation. Table 1 lists physical properties of the reactor coolant and associated estimates of the uncertainties. Table 4 contains design specifications for the reactor coolant. The limitations and conditions imposed by the SE report are related to specific aspects of the KP-FHR design and the need for significant additional testing to gather relevant design and safety data with associated uncertainties related to FLiBe thermophysical properties, corrosion control, and/or reactivity control for the range of conditions required for reactor safety evaluations. The additional testing is to be performed under an approved quality assurance program to validate the properties and design specifications for the range of conditions used in the KP-FHR design and safety analysis. The staff has noted that the results from this testing must be submitted to NRC for review and approval.

Committee Action

The Committee issued a letter dated May 27, 2020, with the following conclusions:

a) The thermophysical properties and design specification limits in Tables 1 and 4 of KP-TR-005-P with the limitations and conditions imposed by the staff SE report provide acceptable initial values for design and safety analyses of the KP-FHR.

b) Limitations and conditions imposed by the staff require an updated version of information in Tables 1 and 4 of KP-TR-005-P be submitted after confirmatory data are obtained under an approved quality assurance program.

c) The SE report should be issued.

d) The proposed Kairos reactor design and limited operational experience with molten salt coolants present several technical issues that could affect either the coolant material properties or the coolant specifications. It is important that information in Tables 1 and 4 be finalized because material properties and coolant specifications are required for acceptance of data obtained from scaled testing using surrogate fluids and are fundamental input to many reactor safety analyses.

3. NuScale Areas of Focus – Probabilistic Risk Assessment (PRA) and Emergency Core Cooling System Valve Performance

As stated in the staff's safety evaluation report, the staff performed their PRA review with a goal to confirm that the Commission guidelines are met. These guidelines include meeting the Commission's Safety Goals, applying PRA to identify and reduce potential design and operational vulnerabilities; and using PRA insights to provide risk-informed support for other programs (e.g., regulatory treatment of non-safety systems, human factors engineering, and the reliability assurance program).

The Committee finds the PRA scope and level of detail sufficient for the discussion of risk results and insights at this stage. However, the risk insights identified in Chapter 19 should not be considered final because there are omissions in the existing FSAR that need to be properly reflected in the PRA.

The risk measures of core damage frequency (CDF) and large release frequency (LRF), quantified in the PRA, would indicate that the NuScale design meets the Commission's Safety Goals with large margins. We agree with the staff conclusion that the low risk estimates "reflect deliberate engineering and design effort to reduce or eliminate the contributors to risk found in previous designs." However, recently identified design issues, underlying omissions, and uncertainties indicate that the large margins between CDF and LRF and safety goals cannot be substantiated at this time.

In addition, as stated in the Chapter 19 safety evaluation report, phenomenological "uncertainties prevented the staff from confirming that the CCFP or deterministic containment performance goals are met." Severe accident simulations, assuming multiple passive cooling system failures, predict that, should the core overheat, core debris would fall into the reactor vessel lower head. The uncertainties for parameters that could affect containment performance in such an event do not support a conclusion that CCFP is sufficiently low. Nevertheless, containment lower head failure from core debris relocation would not lead to a large release because of the scrubbing effect of the pool water.

In our September 25, 2019 letter, the Committee identified the PRA as one of five areas of focus. In our interim review of the PRA, dated June 19, 2019, the Committee concluded that there are technical items in the PRA that merit further consideration to help identify valid risk insights in this unique design. These include: further examination of the design of ECCS valves and the associated PRA model, further investigation of the uncertainty and the sensitivity

analysis, and identification and evaluation of possible errors of commission associated with the RBC operation. Other focus area reviews identified additional issues not addressed in the PRA. One of these issues, the recently identified omission of boron dilution scenarios, could have a high-risk significance. This and other issues are discussed in the Committee's letter dated June 1, 2020.

Committee Action

The Committee issued a letter on this topic dated June 1, 2020, with the following conclusions and recommendations:

a) The NuScale DCA meets the 10 CFR 52.47(a)(27) requirement to include a description of the design-specific PRA and its results in the DCA.

b) A primary purpose of the PRA at the DCA stage is to inform the design to reduce risk. The PRA scope is sufficient to enable the discussion of risk results and insights, and the level of detail in the PRA is consistent with its intended uses in support of design certification; i.e., to identify design alternatives, operational vulnerabilities, and to provide risk-informed support for other programs. However, the risk insights identified in Chapter 19 should not be considered final because there are omissions in the existing Final Safety Analysis Report (FSAR) that need to be properly reflected in the PRA.

c) In our Chapter 15 letter, the Committee identified a boron dilution issue that remains open. We are concerned that this class of events could lead to a potential reactivity insertion accident and core damage. The applicant is working on resolution of this issue. This resolution needs to be evaluated to determine if these scenarios should be included in the PRA at the DCA stage. Such inclusion could impact the reported risk measures and the risk insights as presented in Chapter 19.

d) The risk measures of CDF and LRF, quantified in the PRA, suggest that the NuScale design meets the Commission's Safety Goals with large margins. However, recently identified design issues, underlying omissions, and uncertainties indicate that the large margins between CDF and LRF and safety goals cannot be substantiated at this time.

e) To promote identification of valid risk insights through the Combined License (COL) process, we provide recommendations on several other topics: ECCS valve performance and qualification; risk importance of the chemical and volume control system (CVCS); errors of commission associated with reactor building crane (RBC) operations; risk increase to single unit operation with multiple unit operation and buildout; steam generator integrity; post-accident combustible gas monitoring; and more rigorous treatment of sensitivities and uncertainties.

   The risk insights will be better supported when the COL applicant addresses the requirements of the NuScale design certification rule appendix to 10 CFR Part 52. This includes addressing the COL items, closing the Inspections, Tests, Analyses, and Acceptance Criteria items, updating the site and plant-specific PRA before fuel load, and, of particular interest with respect to the NuScale design, addressing the additional requirements and restrictions in the rule appendix to address design completeness.

f) The Committee cannot reach a final conclusion on the safety of the NuScale design until the issue of the potential for a reactivity insertion accident due to boron dilution in the downcomer is resolved to our satisfaction.

SPECIAL FULL COMMITTEE MEETING IN JULY

In accordance with the ACRS bylaws, Section 1.2, the Committee voted to conduct a special meeting of the Full Committee July 21 – July 23, 2020, with the primary purpose of discussion, review and comment on NuScale issues and letters including the final letter on the NuScale design certification application.

SCHEDULED TOPICS FOR THE 674th ACRS MEETING

The following topics are on the agenda for the 674th ACRS meeting which is scheduled for June 3-5, 2020:

- Letter Writing for Regulatory Guide 1.236, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop"

- Letter Writing for RG 1.187, Guidance for Implementation of 10 CFR 50.59, "Changes, Tests, and Experiments," regarding digital instrumentation and control upgrades

- Continued discussion and letter writing regarding various NuScale design certification issues including boron distribution

Sincerely,

Matthew W. Sunseri Digitally signed by Matthew W. Sunseri
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Matthew W. Sunseri,
Chairman

June 17, 2020

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