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March 18, 2024

Walter L. Kirchner, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESPONSE TO THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS LETTER, "REVIEW OF ADVANCED REACTOR CONTENT OF APPLICATION PROJECT/TECHNOLOGY-INCLUSIVE CONTENT OF APPLICATION PROJECT GUIDANCE"

Dear Chairman Kirchner:

On behalf of the U.S. Nuclear Regulatory Commission (NRC) staff, I would like to thank you for the letter from the Advisory Committee on Reactor Safeguards (ACRS or the Committee) dated December 20, 2023 (Agencywide Documents Access and Management System Accession No. ML23348A182). This letter addressed the effort by the NRC staff and industry to develop guidance for risk-informed, technology-inclusive, non-light-water reactor (non-LWR) applications. I appreciate the substantial time and effort that the ACRS devoted to this subject during numerous subcommittee and full committee meetings, including the most recent ACRS Full Committee meeting held from December 6–7, 2023.

In its letter dated December 20, 2023, the ACRS recognized that the Advanced Reactor Content of Application Project (ARCAP) and Technology-Inclusive Content of Application Project (TICAP) guidance is technology inclusive, risk informed, and performance based and provides flexibility for a range of non-LWR applicants. In addition, the letter states that the guidance "should serve design developers and the staff as a useful starting point to align expectations for the application process and promote high quality submissions."

Conclusions and Recommendations

The NRC staff provides the responses below to the recommendations in the ACRS letter.

Regulatory Guide (RG) 1.253

1. We reiterate the importance of a comprehensive hazard analysis and a thorough accident identification process as being critical for producing a good application. In this context, we urge the staff to expedite conversion of Draft Guide (DG)-1413, "Technology-inclusive Identification of Licensing Events in Commercial Nuclear Power Plants," into an approved regulatory guide so that near-term non-LWR applicants can use this guidance in their Part 50 and 52 applications rather than waiting until Part 53 is finalized.

Staff Response: The NRC staff released a preliminary draft of DG-1413 (RG 1.254) during the development of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 53, "Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors," and discussed the information in the draft guidance during public meetings. The discussions included the applicability of the guidance to prospective applicants under either 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." However, the guidance was prepared such that it would also be applicable to the preliminary proposed 10 CFR Part 53 then in development. The reference to 10 CFR Part 53 in the guidance is the reason that its release for public comment is tied to the rulemaking effort. While the staff is not planning to issue the draft guidance before the Commission issues a proposed 10 CFR Part 53 for comment, the insights offered by DG-1413 have been and will continue to be an important part of the preapplication discussions with prospective applicants.

The staff made no change to DG-1413 or the related plans for issuing it for public comment.

2. Staff guidance does not clarify the relationship among the terms 'fundamental safety functions', 'required safety functions', and 'necessary safety functions'. Nuclear Energy Institute (NEI) 21-07 identified the terms 'fundamental safety functions', 'Probabilistic Risk Assessment (PRA) safety functions', and 'required safety functions'. Staff should define the terms used in their guidance and the need for documentation of the rationale used to identify these functions.

Staff Response: The terms "fundamental safety functions," "required safety functions," and "PRA safety functions," are defined in NEI 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," issued August 2019 (ML19241A336), which was endorsed in RG 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," issued June 2020 (ML20091L698). Those definitions are applicable to the discussions and usage of the terms in NEI 21-07, "Technology Inclusive Guidance for Non-Light Water Reactors: Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology," issued August 2021 (ML21250A378), and RG 1.253 and are provided below for information:

- *Fundamental safety function:* Safety functions common to all reactor technologies and designs; includes control heat generation, control heat removal, and confinement of radioactive material.

- *PRA safety function:* Reactor design-specific structures, systems, and components (SSC) functions modeled in a PRA that serve to prevent and/or mitigate a release of radioactive material or to protect one or more barriers to release. American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS)-Ra-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants," refers to these as "safety functions." NEI 18-04 uses the modifier "PRA" to avoid confusion with safety functions performed by safety-related SSCs.

- *Required safety function:* A PRA safety function that is required to be fulfilled to maintain the consequence of one or more design-basis events or the frequency of one or more high-consequence beyond-design-basis events inside the frequency-consequence target.

In response to the ACRS comment and to avoid confusion from the possible introduction of another term, including the term “safety function,” the staff has removed the phrase “and ensure that necessary safety functions and SSCs are covered under their proposed PDC” from the first paragraph on page 4 of DANU-ISG-2022-01. The updated paragraph is below:

> The NRC staff anticipates non-LWR applicants will review the general design criteria (GDC) and the guidance in RG 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” issued April 2018 (ML17325A611), for appropriate insights to develop their principal design criteria (PDC) ~~and ensure that necessary safety functions and SSCs are covered under their proposed PDC~~. For the applications that follow the risk-informed and performance-based (RIPB) approach in NEI 18-04, Revision 1, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” issued August 2019 (ML19241A336) (called the Licensing Modernization Project (LMP) process), the design-specific criteria identified by the RIPB approach may be used to supplement or modify the applicable GDC or Advanced Reactor Design Criteria in RG 1.232 in the formulation of PDC.

3. Assurance should be provided that the construction permit (CP) PRA results are reasonable (given the maturity of the design) and that the safety analysis report (SAR) provides sufficient information to support the CP findings. The application should also include the commitments to upgrade and maintain the PRA so that its maturity at the operating license (OL) stage is consistent with its intended uses.

Staff Response: The NRC staff acknowledges the comment and notes that the guidance (TICAP RG appendix A) includes such information, including a recommendation that commitments to upgrade and maintain the PRA should be included in CP applications.

The staff has made no changes to DANU-ISG-2022-01 or RG 1.253.

4. There is insufficient guidance in this RG or NEI 21-07 on how the evaluation of cliff edge effects is to be documented in the SAR. The LMP methodology requires determining whether cliff edge effects are present, and other documents (such as RG 1.242) require addressing cliff edge effects as part of the overall plant design. However, as discussed in INL/EXT-20-60392, the LMP does not specifically identify a methodology to determine whether a specific non-LWR design exhibits cliff edge effects. INL/EXT-20-60392 notes that the method recommended by the International Atomic Energy Agency (IAEA) in SSG-2 for identifying cliff edge effects is, “the performance of sensitivity studies using the deterministic models for evaluating the plant response to events.” The staff should consider this IAEA guidance or comparable guidance. Furthermore, a subsection of the appropriate SAR chapter should be identified for discussion of both the process and the results for cliff edge determination.

Staff Response: The staff agrees that the evaluation of potential cliff-edge effects is an important part of the LMP methodology and should be discussed in the SAR. The need to evaluate cliff-edge effects is addressed in NEI 18-04, RG 1.233, and ASME/ANS-RA-S-1.4-2021 in terms of the risk assessment supporting the LMP methodology. In response to the ACRS comment, the staff has added the following paragraph to Section C.2, "Staff Position," of RG 1.253:

> *(d) Addition: NEI 21-07 calls for the application to include discussions of the analyses of the potential radiological consequences from various event sequences identified from the PRA and related assessments. NEI 18-04 includes a specific question to be addressed during the integrated decision-making process related to the assessment of "cliff edge effects." ASME/ANS-RA-S-1.4-2021 addresses possible cliff-edge effects in areas such as seismic and flooding hazards. Any design requirements or special treatment of SSCs to prevent or mitigate cliff-edge effects should be included in the SSC-specific descriptions in subsequent SAR chapters.*

Further, RG 1.242, "Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities," issued November 2023 (ML23226A036), also addresses cliff-edge effects in terms of consequence assessments for consideration in establishing emergency planning zones under 10 CFR 50.160, "Emergency preparedness for small modular reactors, non-light-water reactors, and non-power production or utilization facilities." However, RG 1.242 is guidance for the operational program of emergency preparedness (EP) and does not contain guidance for plant design. This does not mean that designers should not consider EP in their designs, only that the NRC's EP regulations and associated guidance, including those for EP zones, are not design requirements. As an independent final layer of defense in depth, EP deals with all residual risks of a facility and relies on information provided within the entire licensing basis, not just the design, to scope the planning efforts.

DANU-ISG-2022-005, ARCAP Chapter 11

1. Rather than relying heavily on NuScale as a precedent, guidance should emphasize the factors noted in our May 21, 2021, letter on control room staffing. These include relevant design details, passive or inherent safety features, accident progression timing, safety margins, the reliance on operator intervention, a thorough safety test program, and a robust operator training and simulator validation program.

Staff Response: ARCAP Chapter 11 contains guidance for both applicants and NRC staff reviewers. The "Staff Review Guidance" portion of Section 11.1.2, "Basis/number of Operating Shift Crews, their Staffing, and Responsibilities," directs the NRC staff to use NUREG-1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)," issued July 2005 (ML052080125), to review requests for exemptions from licensed operator staffing requirements. NUREG-1791 provides a comprehensive human factors engineering-based framework for the evaluation of staffing plans, and its methodology bounds the considerations noted in the Committee's letter dated May 21, 2021. Specifically, the NUREG-1791 process considers, in part, the concept of operations (of which the training required for control personnel is a portion), operational conditions, function allocation, task analysis, human-system interface design, and the staffing plan with its supporting validation. The NuScale example is used within the "Application

Guidance" portion of section 11.1.2 to illustrate how design certifications offer an alternative regulatory pathway for implementing the modified staffing model using an approach based on NUREG-1791. For these reasons, the staff believes that the existing ARCAP chapter 11 content captures the substance of the Committee's comment in this area.

The staff made no change to this ISG.

DANU-ISG-2022-006, ARCAP Chapter 12

1. To make this guidance [Post-construction Inspection, Testing, and Analysis Program (PITAP)] more complete, consider the following: The lack of any guidance on building environmental conditions (heating, cooling, ventilation, lighting etc.) is notable considering that new reactors may be sited in harsh and remote environments. Building environmental conditions should be explicitly noted to remind the reviewer to assess unusual environmental conditions that may impact SSCs for a given reactor design.

Staff Response: The staff agrees that inspections, testing, and analyses need to confirm safety-related and non-safety-related with special treatment SSC performance over the range of site-specific environmental conditions. Building environmental condition considerations are addressed through a combination of guidance associated with meeting the requirements for the design of safety-related and non-safety-related with special treatment SSCs, siting of reactors (including the identification of various site characteristics), assessing human factors considerations, and meeting the acceptance criteria identified in DANU-ISG-2022-006. While many different factors influence design, operator actions, and programmatic controls, the staff does not believe it is necessary to include specific examples throughout the guidance documents or to emphasize one area over others by providing isolated examples. To the extent the Committee is suggesting that the ARCAP guidance needs to instruct NRC staff reviewers to consider whether SSCs are designed to withstand or are protected from external phenomena at the proposed location of the facility, DANU-ISG-2022-002 inherently includes such guidance.

The staff made no change to this ISG.

2. To make this guidance [PITAP] more complete, consider the following: The retention and organization of records is vital for efficient and effective NRC audits and inspections prior to final issuance of an OL, manufacturing license, or combined operating license (COL). The ISG mentions test reports and the need for NRC inspections, however, the need to keep all records organized and complete, should be emphasized.

Staff Response: The staff agrees that it is important to both retain records and maintain the ability to retrieve them to support the activities of both applicants and licensees on the one hand and the NRC staff on the other. However, the regulatory requirements for record retention and the records management systems currently in use by applicants and licensees adequately address the need to retrieve the records to support efficient and effective NRC audits and inspections during the construction period.

The staff made no change to this ISG.

DANU-ISG-2022-007, ARCAP Inservice Inspection (ISI)/Inservice Testing (IST)

1. Much was learned about the behavior of materials used in LWRs even after decades of operation. The guidance should retain an element of nimbleness to adapt to lessons learned and operating experience that will be gained as new non-LWRs develop their own unique operating experience.

Staff Response: In response to the ACRS comment, the staff added the following to DANU-ISG-2022-007, “ARCAP Risk-informed Inservice Inspection and Inservice Testing,” as a new paragraph after “Use of Risk Information” on page 5 of the ISG:

> *Behavior of Materials*
>
> *The available performance data for materials used for SSCs designated as safety-related and non-safety-related with special treatment for novel designs, or technologies might be limited; therefore, an OL or COL applicant will need to describe the ISI/IST measures and associated bases for those SSCs in the application. The development of ISI/IST measures for non-LWR designs may incorporate performance-based concepts similar to those developed for LWRs. Such performance-based approaches can address the limited operating experience that may be initially available for some designs and materials. These approaches might identify appropriate adjustments in the types and frequency of inspections and testing as operating experience is gained over the subsequent years and decades. Some consensus standards include provisions for the evaluation of performance information, such as ASME BPV Code, Section XI, Division 2, “Reliability Integrity Management (RIM) Program,” which requires reevaluation of the RIM program based on new information such as changes in SSC performance, new service-related degradation, and industry operating experience.*
>
> *In addition, the ASME is revising ASME Standard QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” to allow its more effective use for the qualification of components to be used in non-LWRs. The qualification process for components in non-LWRs will include the consideration of materials to be used in components in advanced reactors. When the ASME issues the revised QME-1 Standard, the NRC staff will consider whether to endorse it in RG 1.100, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants” or other appropriate guidance. During plant operations, the licensee should evaluate operating experience to determine if the unique operating experience of non-LWRs indicates the need for attention regarding the original qualification of materials used in non-LWRs. In addition, ISI/IST programs for non-LWRs might need to be updated based on lessons learned from SSC performance, service-related degradation, and operating experience.*

2. Concepts that are new, such as “Components that Control Fluid without Mechanically Interacting with the Fluid”, should be a focus for industry interaction to ensure alignment on expectations for license applications.

Staff Response: The staff agrees with the ACRS that new and novel design features will present challenges during the development and review of both reactor designs and associated programs for inspections and testing.

In response to this comment, the staff has added further discussion at the end of "Components that Control Fluid without Mechanically Interacting with the Fluid" on page 4 of the ISG:

> *Novel concepts for non-LWRs, such as components that control fluid without mechanically interacting with the fluid, should be a focus of preapplication interactions with the NRC staff. Such interactions ensure the NRC staff understands the proposed design and the prospective applicant understands the level of detail needed to describe the novel concepts in the license application. For such interactions, a prospective applicant should consider the guidance associated with the design of safety-related or non-safety-related with special treatment SSCs (including DANU-ISG-2022-01 (Roadmap)), the inspections and testing covered by DANU-ISG-2022-007, and other available information (including consensus codes and standards). The prospective applicant should also consider any new IST codes for non-LWRs that recognize the potential use of components that move fluid or stop fluid from moving in non-LWRs based on design principles that significantly differ from those used to design typical pumps and valves used in currently operating nuclear power plants.*

3. Documents referenced in the body of the document, such as NEI 18-04, should be added to the reference section.

Staff Response: In response to the ACRS comment, the staff added references to DANU-ISG-2022-007 for documents cited in the body of the guidance.

DANU-ISG-2022-009, ARCAP Fire Protection

1. The approach proposed in this ISG practically suggests that the "voluntary" pre-application process may be largely mandatory because several key assumptions (such as use of NFPA codes not endorsed by the NRC) must be vetted through the NRC staff prior to expenditure of significant applicant/designer resource to design an acceptable program.

Staff Response: The NRC staff has included Appendix A, "Pre-Application Engagement Guidance," to DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," to encourage and support preapplication interactions. While NRC regulations do not require such interactions, the staff does not dispute the Committee's observation that applications not supported by preapplication activities may incur longer review times and costs. The staff continues to encourage applicants to identify and, where possible, resolve key technical issues through topical reports or endorsement of industry codes and standards before submitting applications for licenses, certifications, or approvals.

In response to the ACRS comment, the NRC staff made the following change to Section A.6.i of DANU-ISG-2022-009, "ARCAP Fire Protection for Operations," to note that preapplication discussions are encouraged, but not required:

> 6. NFPA 804, "Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants" (Ref. 11)
>
> i. provides useful information when used in conjunction with NRC regulations and guidance—the NRC has not formally endorsed NFPA 804, and some of the information in the NFPA standard may not comply with regulatory requirements

and may accordingly require an exemption and possibly additional action if utilized. Applicants ~~should~~ *are encouraged to* discuss their use of NFPA 804 with NRC staff during preapplication interactions.

2. Furthermore, the ISG assumes the advanced non-light water reactor site will have a fire brigade and is silent on any extension of qualifications, training, or agreements with offsite resources. Not all non-LWR plant organizations will have a fire brigade, essentially relying on offsite resources. The ISG is silent on guidance for the situations where the applicant opts for primary reliance on offsite fire response. Guidance should be developed, or existing guidance referenced to ensure the emergency planning and offsite coordination is appropriate for an operational fire protection program.

Staff Response: In response to the ACRS comment, the staff added the following note to the guidance on page 4 of DANU-ISG-2022-009:

The fire protection program description should also describe specific features necessary to implement the program, such as the following:

1. administrative controls and personnel requirements for fire prevention and manual fire suppression activities*

2. the means to limit fire damage to safety-related or non-safety related with special treatment SSCs so their capability to perform their functions is maintained

**This ISG provides one acceptable approach to meeting regulatory requirements and was developed from guidance for large LWRs, including measures such as fire brigades to provide timely manual fire suppression responses. Applicants considering fire protection programs that do not rely on these measures (for example, an onsite fire brigade) should demonstrate their ability to safely shut down the facility and minimize radioactive releases to the environment in the event of a fire without the excluded measures. The NRC staff encourages preapplication discussions for facilities that primarily rely on offsite fire response. For LMP-based plants, how fires are addressed in the full-scope PRA could be part of the discussion because the response of the facility to fires could be a part of the justification for the primary reliance on offsite fire response.*

In addition, RG 1.242, "Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities," issued November 2023 (ML23226A036), includes guidance for developing applicant emergency plans for offsite fire response organizations that may have onsite responsibilities. The RG includes development of procedures for notifying, training, and instituting drills for offsite fire response organizations that include the use of the fire suppression technology. RG 1.242 applies whether or not the applicant takes credit for an onsite fire brigade.

ARCAP Roadmap

1. Other Comments. ARCAP guidance and the suite of ISGs do not contain planning requirements for decommissioning. In order to establish adequate financial qualifications to construct and operate the facility, at a

minimum, a high-level decommissioning strategy should be described in the SAR.

Staff Response: In response to the ACRS comment, the staff added the following sentences to DANU-ISG-2022-001:

Financial Qualification—Application Guidance

RG 1.206, Section C.1.1, “General and Financial Information,” Revision 1, provides application content guidance that generally applies to LWR and non-LWR technologies. Additional guidance can be found in RG 1.159, “Assuring the Availability of Funds for Decommissioning Nuclear Reactors,” Revision 2, issued October 2011, and NUREG-1577, “Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance,” Revision 1, issued February 1999. The staff notes that the decommissioning guidance found in these guidance documents is based on LWR technologies. *The cost estimates for decommissioning included in the application must be developed considering costs in such areas as engineering, labor, and waste disposal. The derivation of the generic cost estimates for LWRs in 10 CFR 50.75 is provided in NUREG/CR-5884, “Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station,” and NUREG/CR-6187, “Revised Analyses of Decommissioning for the Reference Boiling Water Reactor Power Station.”* Non-LWR applicants are encouraged during the preapplication phase to discuss with the staff their plans for developing decommissioning funding estimates for their specific non-LWR technology.

The NRC staff appreciates the continued engagement of the ACRS on the ARCAP/TICAP guidance and considers the Committee’s recommendations to be valuable input to this complex guidance development effort.

Sincerely,

Andrea Veil Signed by Veil, Andrea on 03/18/24

Andrea D. Veil, Director
Office of Nuclear Reactor Regulation

cc: Chair Hanson
Commissioner Wright
Commissioner Caputo
Commissioner Crowell
SECY

SUBJECT: RESPONSE TO THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS LETTER, "REVIEW OF ADVANCED REACTOR CONTENT OF APPLICATION PROJECT/TECHNOLOGY-INCLUSIVE CONTENT OF APPLICATION PROJECT GUIDANCE," DATED: MARCH 18, 2024

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