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General Comment

Please see attached comment from SMR, LLC.

Attachments

DANU-ISG-2022-01 Public Comment SMR LLC

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August 10, 2023

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Subject: SMR, LLC Comment on DANU-ISG-2022-01; Docket ID NRC-2022-0074

SMR, LLC (SMR), a wholly owned subsidiary of Holtec International, appreciates the opportunity to publicly comment on the DANU-ISG-2022-01 draft interim staff guidance "Review of Risk-Informed, Technology-Inclusive Advanced Reactors Applications–Roadmap" issued in May 2023. This comment focuses on the extent of transient and accident analysis methodologies and the underpinning code validation and verification required in a Preliminary Safety Analysis Report (PSAR) for a construction permit application (CPA). The relevant regulations within 10 CFR 50 appear to be as follows:

- 10 CFR 50.34(a), which states, in part [<u>emphasis added</u>]:
  (a) Preliminary safety analysis report*. <u>Each application for a construction permit shall include a preliminary safety analysis report. The minimum information to be included shall consist of the following</u>:*

  (4) A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. <u>Analysis and evaluation of ECCS cooling performance and the need for high point vents following postulated loss-of-coolant accidents must be performed in accordance with the requirements of § 50.46 and § 50.46a of this part for facilities for which construction permits may be issued after December 28, 1974.</u>

- 10 CFR 50.46(a)(1), which states [<u>emphasis added</u>]:
  (a)(1)(i) Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. <u>Except as provided in paragraph (a)(1)(ii) of this section, the</u>

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> *evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model.* *This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted.*
>
> *(ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models.*

- 10 CFR Appendix K, II. Required Documentation, which states, in part:
  5. General Standards for Acceptability—Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including: For models covered by § 50.46(a)(1)(ii), compliance with required features of section I of this Appendix K; and, for models covered by § 50.46(a)(1)(i), assurance of a high level of probability that the performance criteria of § 50.46(b) would not be exceeded.

DANU-ISG-2022-01 was developed as part of the Advanced Reactor Content of Applications Project (ARCAP). Per the NRC ARCAP webpage, "The purpose of the [ARCAP] is to develop technology-inclusive, risk-informed and performance-based guidance." SMR understands this effort was initiated to develop guidance for non-light water reactor (non-LWR) applications. However, SMR believes that "technology-inclusive" guidance is also intended to apply to LWRs.

Appendix C of DANU-ISG-2022-01, which provides construction permit (CP) guidance, provides two portions of guidance. The first, which is stated as applicable to both LWRs and non-LWRs, directly incorporates portions of DRNL-ISG-2022-01 "Safety Review of Light-Water Power-Reactor Construction Permit Applications" issued in October 2022. The second, which is stated as only applicable to non-LWRs, details new guidance. Of particular interest in the second section of DANU-ISG-2022-01 Appendix C is the discussion on Safety and Accident Analysis Methodologies and Associated Validation, which states:

"*Construction permit applicants should develop and execute plans to perform safety and accident analyses that include testing of safety features to support validation and verification of associated engineering computer programs. The approval of these analysis plans needs to include development of associated methodologies and applications of those methods, which include but are not limited to event-specific*

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analysis methodologies, scaling methodology, setpoint methodology, reactor coolant analysis methodology, core design methodology, and reactivity control methods. The analysis plans need to include a test plan and test program to ensure appropriate verification and validation of the engineering computer programs, including consideration of appropriate quality assurance requirements. The test program should satisfy 10 CFR 50.43(e), which requires applicants to demonstrate that sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions."

This is in contrast with specific portions of DRNL-ISG-2022-01 that discuss evaluation methodologies, which are not incorporated into DANU-ISG-2022-01 Appendix C. Specifically, DRNL-ISG-2022-01 states, in part:

"*At a minimum, the NRC staff should ensure the preliminary safety analysis report includes all the information required by 10 CFR 50.34, with a focus on the following:*
…
- *Verification that the LOCA evaluation methods used are approved and applicable to the design.*
- *Verification that non-LOCA evaluation methods are at a minimum under active NRC staff review and any open items can reasonably be left for later consideration in the final safety analysis report, and that there is reasonable assurance that the proposed facility can be constructed and operated without undue risk to public health and safety.*"

SMR believes that the DANU-ISG-2022-01 Appendix C guidance specific to evaluation models for non-LWRs differs from the DNRL-ISG-2022-01 guidance specific to evaluation models for LWRs due to the assumption that 10 CFR 50.46 does not apply to non-LWRs, and therefore the expectation that code V&V and evaluation models be approved in the PSAR does not extend to non-LWRs. This is further evidenced by the following NRC discussion in the Safety Evaluation for Hermes Construction Permit Application (ML23158A268):

- Section 4.5.3.1: "*While the staff reviewed uncertainties in [nuclear design] models, the staff did not make any findings on Kairos's validation and verification plan of codes or derivations of uncertainties because it is not required or necessary for the issuance of a CP. Kairos's validation and verification of codes and derivations of uncertainties will be reviewed during the OL application.*"

- Section 13.2.2.4: *"The staff finds that the level of detail provided on the postulated event evaluation methodology is consistent with the applicable guidance and acceptance criteria in NUREG-1537, Parts 1 and 2, Section 13, "Accident Analysis" and demonstrates an adequate design basis for a preliminary 13-43 design. Based on the technical evaluation discussed above, the staff finds that the methodology used to show that the radiological consequences of the postulated events are bounded by the MHA analysis is sufficient and that the*

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methodology meets the applicable regulatory requirements and guidance identified in this section for the issuance of a CP in accordance with 10 CFR 50.35 and 50.40. Further technical or design information required to complete the safety analysis may reasonably be left for later consideration. The staff will confirm that the final design conforms to the design basis during the evaluation of the FSAR as part of the OL application review."

- Section 4.1.2 of technical report KP-TR-018-NP (ML21272A38), which is describes the postulated event evaluation methodology and is referenced in the Hermes PSAR, states, in part, "*[The postulated event evaluation methodology] will be verified and validated prior to the final safety analysis.*"

Though there appears to be a valid regulatory basis for not requiring approved evaluation methodologies in a non-LWR PSAR, SMR notes the juxtaposition of holding LWRs, with substantial operating experience in the US, to a more stringent standard compared to non-LWRs, which have limited operating experience in the US. SMR requests clarity if this understanding is consistent with the NRC's interpretation of the pertinent regulations and the intended message of the DANU-ISG-2022-01 and DRNL-2022-01 draft guidance. If SMR understands the current regulations and available interim staff guidance documents correctly, SMR encourages the NRC to consider pathways to similarly reduce the threshold of transient and accident analysis methodologies and the underpinning code validation and verification necessary to accept and approve a LWR construction permit application, consistent with that of a non-LWR.

Respectfully,

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