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| **From:** | Carolyn Lauron |
| **To:** | Justin Hawkins; Andrew Brenner |
| **Cc:** | Manny Sayoc; Michelle Hayes; Greg Cranston |
| **Subject:** | NRC Staff Questions and SMR (Holtec) Responses re: July 19, 2023 Public Meeting Materials - SMR-160 Code Verification and Validation and Evaluation Methodologies (99902049) |
| **Date:** | Tuesday, July 25, 2023 8:27:00 AM |

Hi Justin, Andrew –

The purpose of this email is to document the NRC staff questions and SMR (Holtec) responses regarding the Presentation Materials for the July 19, 2023, Public Meeting on SMR-160 Code Verification and Validation and Evaluation Methodologies.

The NRC staff questions were emailed on July 17, 2023, and the SMR (Holtec) responses were received on July 19, 2023.

Thank you,
Carolyn Lauron
US NRC

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**NRC Staff Questions:**

- Slide 6 – What does representative calculations mean?
- Slide 9 – Applicable to other codes in the presentation – which ones?

**SMR (Holtec) Response:**

**SMR prepared the following response, in part, to answer the NRC staff's question "What does 'representative calculations' mean" in the context of slide 6. SMR also intends to use this response to document the current understanding of various regulations and guidance as it applies to the presentation topic.**

The current best estimate schedule for submitting LTRs documenting code V&V and evaluation methodologies for performing the final safety analysis of SMR-160 extends beyond the targeted SMR-160s at Palisades CPA submittal date. If acceptable to the NRC, SMR intends to submit Chapter 15 of the PSAR that is based on "representative calculations" of each limiting AOO and DBA performed by codes that have not completed the EMDAP process in RG 1.203. This is in contrast of the Chapter 15 analysis required in the FSAR, for either an OLA or Part 52 application, that would be based on submitted (if not already approved) LTRs. Since the intended SMR-160 PSAR Chapter 15 would be based on these "representative calculations" SMR plans to only perform a single calculation for each major design basis event, using engineering judgement to determine inputs and assumptions that are judged to result in close-to-limiting conditions rather than analyzing a full matrix of initial conditions, single failures, and treatment of non-safety systems for each event to ensure the bounding scenario is captured in the PSAR. This approach is being considered given this broad matrix would need to be re-analyzed once code V&V is complete and evaluation methods are finalized. It is SMR's belief that performing these exhaustive analyses in the PSAR seems to add little value given the expectation that the bounding cases, as will be identified in the FSAR by V&V'd codes and final evaluation methodologies, are not expected be substantially different than results presented in the

PSAR. As evidenced by the presentation, and which will remain evident in future preapplication engagements that lead up to a submitted CPA, SMR intends for the LTRs to be completed well in advance of construction completion and OLA submittal.

However, this approach may be inconsistent with relevant regulations, specifically:

- 10 CFR 50.34(a), which states, in part [emphasis added]:

  *(a)* Preliminary safety analysis report. *Each application for a construction permit shall include a preliminary safety analysis report. The minimum information to be included shall consist of the following:*

  *(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. 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.*

- 10 CFR 50.46(a)(1), which states [emphasis added]:

  *(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. Except as provided in paragraph (a)(1)(ii) of this section, the 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, 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.*

DNRL-ISG-2022-01 (ML22189A099) "Safety Review of Light-Water Power Reactor Construction Permits" provides additional guidance as it relates to evaluation methods, as it 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.*"

The above regulations and guidance are consistent with all communication on this topic that SMR has received from the NRC to-date. However, SMR has subsequently noted 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 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."*

SMR has reviewed NUREG-1537 Parts 1 and 2, Section 4, "Reactor Description" and Section 13, "Accident Analysis" to understand if there are differences between guidance for

PSAR and FSAR content for accident evaluation methodologies of non-power test reactors when compared to that of light-water power reactors as provided in the appropriate sections of NUREG-0800. Neither NUREG-1537 nor NUREG-0800 appears to differentiate guidance for a PSAR and an FSAR in discussing accident evaluation methodologies. However, this Kairos Power approach, as approved by the NRC, is consistent with DANU-ISG-2022-01 (ML22048B546) "Review of Risk-Informed, Technology Inclusive Advanced Reactor Applications-Roadmap" Appendix C "Construction Permit Guidance". Appendix C is broken down into two parts, one which incorporates excerpts from DNRL-ISG-2022-01 that includes guidance applicable to both LWRs and non-LWRs, and another that is only applicable to non-LWRs. The DRNL-ISG-2022-01 section quoted above is not included in DANU-ISG-2022-01 Appendix C, but the section including guidance specific to non-LWRs states, in part:

"*Safety and Accident Analysis Methodologies and Associated Validation*

*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 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.*"

SMR believes this guidance is consistent with the NRC discussion in the Safety Evaluation for Hermes Construction Permit Application given the Hermes reactor is a non-LWR.

SMR further 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 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. Though there appears to be a valid regulatory basis for this conclusion, SMR notes it is challenging to accept that LWRs, which have a substantial amount of operating experience in the US, are required to be held to a higher standard during a CPA review of preliminary design information than non-LWRs, which have significantly less operating experience in the US.

SMR also notes 10 CFR 50.35, which states [<u>emphasis added</u>]:

*(a) <u>When an applicant has not supplied initially all of the technical information required to complete the application and support the issuance of a construction permit which approves all proposed design features, the Commission may issue a construction permit if the Commission finds that</u> (1) the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the*

*protection of the health and safety of the public; (2) such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the final safety analysis report; (3) safety features or components, if any, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components; and that (4) on the basis of the foregoing, there is reasonable assurance that, (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in part 100 of this chapter, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.*
*Note: When an applicant has supplied initially all of the technical information required to complete the application, including the final design of the facility, the findings required above will be appropriately modified to reflect that fact.*

*(b) A construction permit will constitute an authorization to the applicant to proceed with construction but will not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated in the permit. The applicant, at his option, may request such approvals in the construction permit or, from time to time, by amendment of his construction permit. The Commission may, in its discretion, incorporate in any construction permit provisions requiring the applicant to furnish periodic reports of the progress and results of research and development programs designed to resolve safety questions.*

*(c) Any construction permit will be subject to the limitation that a license authorizing operation of the facility will not be issued by the Commission until (1) the applicant has submitted to the Commission, by amendment to the application, the complete final safety analysis report, portions of which may be submitted and evaluated from time to time, and (2) the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility in accordance with the requirements of the license and the regulations in this chapter.*

SMR hopes the above response can facilitate discussion with the NRC staff on what is needed in terms of code V&V and evaluation methodologies for CPA submittal and CPA review, and whether 10 CFR 50.35 allows for any leniency in the requirements of 10 CFR 50.34(a)(4).