UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 1, 2021

MEMORANDUM TO: Michael I. Dudek, Branch Chief
New Reactor Licensing Branch
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation

FROM: Gregory V. Cranston, Project Manager ***/RA/***
New Reactor Licensing Branch
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation

SUBJECT: SUMMARY OF THE JUNE 16, 2021, PUBLIC MEETING TO DISCUSS THE HOLTEC SMALL MODULAR REACTOR, SMR-160, REQUEST FOR ADDITIONAL INFORMATION ON LICENSING TOPICAL REPORT: "ELIMINATION OF LARGE BREAK LOSS-OF-COOLANT ACCIDENT AND ESTABLISHMENT OF LOCA ACCEPTANCE CRITERIA"

On June 16, 2021, an Observation Public Meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and SMR, LLC, a Holtec International Company (Holtec), regarding NRC staff's request for additional information regarding licensing topical report (LTR) HI-2201064R2, "Elimination of Large Break Loss-of-Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria," (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML21068A255). The meeting summary is provided in Enclosure (3). The public meeting notice can be found in ADAMS under Accession No. ML21158A285 and was also posted on the NRC's public Web site.

Enclosed are the meeting agenda (Enclosure 1), list of attendees (Enclosure 2), meeting summary (Enclosure 3), and an attachment identifying supplemental NRC staff comments for items mentioned but not discussed in detail during the meeting (Enclosure 4). Enclosure 4 supplements Enclosure 3 in identifying where additional information should be provided by Holtec to supplement their current response (ADAMS Accession No. ML21147A532) to the request for additional information (ADAMS Accession No. ML21123A187). This information in Enclosures 3 and 4 can be discussed with SMR-LLC, if clarification is needed, at a future public meeting.

Docket No. 99902049

Enclosures:
1. Meeting Agenda
2. List of Attendees
3. Meeting Summary
4. Supplemental Meeting Comments

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DATED: JULY 1 2021

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ADAMS Accession Nos:
PKG: ML21180A466
MEMO: ML21180A465
MEETING NOTICE: ML21158A285 *** via e-mail** **NRR-106**

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SUMMARY OF THE JUNE 16, 2021, PUBLIC MEETING TO DISCUSS THE HOLTEC SMALL MODULAR REACTOR SMR-160 REQUEST FOR ADDITIONAL INFORMATION ON LICENSING TOPICAL REPORT: "ELIMINATION OF LARGE BREAK LOSS-OF-COOLANT ACCIDENT AND ESTABLISHMENT OF LOCA ACCEPTANCE CRITERIA"

June 16, 2021

Meeting Agenda

Time	**Topic**	**Organization**
1:00 p.m. – 1:10 p.m.	Introductions and Opening Remarks	NRC and Holtec
1:10 p.m. – 1:30 p.m.	SMR-160 Licensing Topical Report Discussion – Open Session	NRC and Holtec
1:30 p.m.	Adjourn	

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June 16, 2021

List of Participants

Name	Affiliation
Joseph Rajkumar	Holtec
Kevin Hickey	Holtec
Rick Trotta	Holtec
Tammy Morin	Holtec
Thomas Marcille	Holtec
Barrett, Antonio	NRC
Brown, Christopher	NRC
Buford, Angie	NRC
Caldwell, Bob	NRC
Cranston, Greg	NRC
Dudek, Michael	NRC
Grady, Anne-Marie	NRC
Honcharik, John	NRC
Hsu, Kaihwa	NRC
Li, Yueh-Li	NRC
Manoly, Kamal	NRC
Mitchell, Matthew	NRC
Nolan, Ryan	NRC
Patton, Rebecca	NRC
Sugrue, Rosemary	NRC
Sweeney, Zach	NRC
Tsao, John	NRC
Tseng, Ian	NRC
Tsirigotis, Alexander	NRC
Villarreal, Tristan	NRC
Wittick, Brian	NRC

SUMMARY OF THE JUNE 16, 2021, PUBLIC MEETING TO DISCUSS THE HOLTEC SMALL MODULAR REACTOR SMR-160 REQUEST FOR ADDITIONAL INFORMATION ON LICENSING TOPICAL REPORT: "ELIMINATION OF LARGE BREAK LOSS-OF-COOLANT ACCIDENT AND ESTABLISHMENT OF LOCA ACCEPTANCE CRITERIA"

June 16, 2021

Meeting Summary

On June 16, 2021, an Observation Public Meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and Holtec regarding NRC's staff request for additional information regarding licensing topical report HI-2201064R2, "Elimination of Large Break Loss-of-Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria," (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML21064A037). The public meeting notice can be found in ADAMS under Accession No. ML21158A285 and was also posted on the NRC's public Web site.

Holtec is requesting that postulation of a break of the steam generator (SG) to reactor pressure vessel (RPV) forged connection (planar inter vessel forging or PIF) and the SG riser be excluded from design-basis LOCA considerations for the SMR-160 such that any breaks associated with the forged connection and SG riser be considered a beyond design basis event. The SG is connected directly to the RPV by a single forging with concentric fluid flow paths. The forged connection goes from the RPV to the SG bottom tubesheet. The SG riser extends from the bottom tubesheet to the top tubesheet is welded to both tubesheets and is continuously supported and guided throughout its length. Coolant heated by the core (hot leg) flows through the inner duct of the PIF to the SG tubesheet and coolant returning to the RPV flows through the outer annulus of the PIF (cold leg).

The purpose of the meeting was to discuss the response that Holtec provided to the NRC staff's request for additional information (RAI-9832, ML21147A532) and to clarify what additional information is required from Holtec in their response to the RAI in order for NRC staff to be able to accept the response and close the RAI as resolved. The focus of the meeting was to obtain more technical information regarding (1) the forged connection between the SG and RPV and (2) the SG riser; and to provide feedback to Holtec as to what additional information is needed.

The public meeting commenced with opening remarks and an introduction of participants. There were no public participants and there were no public comments.

During the public portion of the meeting, the NRC staff discussed the additional quantitative information that is needed to better understand Holtec's basis for excluding LOCA considerations for specific reactor locations. The RAI response did not provide sufficient details and specificity regarding the design, construction, inspection, operation, and monitoring of the forged connection and associated welds to be able to determine that the probability of a break would be sufficiently low such that the NRC can conclude the forged connection, including the riser, can be excluded from LOCA considerations.

Since there were no members of the public in attendance, the meeting transitioned to the closed portion so that proprietary information could be discussed.

At the beginning of the closed portion of the meeting, the NRC staff reiterated that additional information will be required to determine whether the vessel to vessel forged connection and the

SG riser can be removed from LOCA consideration to determine what limitations or conditions might need to be applied to the topical report and if exemptions from the NRC regulations are needed.

The key discussion points requiring additional information are listed below and additional information on these discussion points is included in Enclosure (4) of this meeting summary.

The NRC staff noted that additional information is needed in the following areas:

- Defense-in-depth considerations if the subject locations fail.
- Types and sizes of breaks that are considered within the design-basis for the subject locations.
- Safety significance related to the combined vessel design and configuration.
- Forged connection and riser welds, such as materials, inspections, fabrication and resulting residual stresses, environmental and operating conditions, and their stress state to evaluate whether certain welds may be excluded from the large break LOCA analysis.
- Weld accessibility for installation and preservice and inservice inspections, including inspection methods, frequency, and acceptance criteria.
- Potential for stress corrosion cracking, thermal (aging) embrittlement, and hydrogen embrittlement and mitigation for these phenomena.
- Descriptions of proposed deterministic and probabilistic fracture mechanic analyses (methodology and acceptance criteria) to be used to demonstrate that the forged connection and SG riser have a sufficiently low probability of failure.
- Loads associated with water hammer, steam hammer and flow induced vibration

See Enclosure (4) for more specific details on the above subjects and additional staff observations related to the RAI response. Also, follow-up discussions on these technical and regulatory issues will be conducted with Holtec later during public meetings and could be the subject of a regulatory audit or an additional RAI.

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June 16, 2021

Supplemental Meeting Comments

In its Licensing Topical Report (LTR), "Elimination of Large Break Loss-of-Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria," Holtec is requesting that postulation of a LOCA of the steam generator (SG) to reactor pressure vessel (RPV) forged connection (planar inter vessel forging or PIF), and the associated SG riser, be excluded from the SMR-160 small modular reactor (SMR) design basis. In the request for additional information (RAI-9832, ML21147A532), the NRC requested that Holtec provide justification that their proposed acceptance criteria support their conclusion that a large break LOCA does not need to be assumed as a design basis accident with respect to 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR Part 50, Appendix A, GDC 35, "Emergency Core Cooling," requirements. The RAI is related to the acceptance criteria, design, and mitigation features and associated commitments proposed as a basis to eliminate a design basis LOCA at the forged connection and at the SG riser.

The staff reviewed the response to RAI-9832 with Holtec, submitted on May 27, 2021. The RAI response generally expresses the same approach that the staff has already communicated would not satisfy the regulatory requirements of 10 CFR 50.46, GDC 35, and 10 CFR Part 100, "Reactor Site Criteria." Specifically, the response does not adequately address the applicability of 10 CFR 50.46 and the need for an applicant seeking to design or construct the SMR-160 to request an exemption to exclude consideration of breaks in the combined vessel from the design-basis. Neither does the response address GDC 35 which also requires the consideration of breaks in the reactor coolant pressure boundary (RCPB). The staff notes that the RAI response did not result in any changes to the LTR.

The below discussion identifies examples of information the staff needs to continue its review of the subject LTR. As stated in the RAI, the topical report should be updated accordingly to address the items below and align with applicable regulations.

<u>Regulatory Applicability, Defense-in-Depth, and Mitigation Capability</u>

The requirements in 10 CFR 50.46 are applicable if either (1) the locations are considered pipes for purposes of 10 CFR 50.46 (note this is independent from the applied American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code) and designation of the locations), or (2) the locations are not considered pipes, but warrant special considerations for inclusion in 10 CFR 50.46 based on special safety significance related to the combined vessel design and configuration. The subject locations primarily act as conduits to transfer fluid, therefore, rendering their consideration for the purpose of 10 CFR 50.46 applicability as pipes. However, even if the staff did agree that the locations were part of a combined vessel the staff would consider whether there was a matter of special safety significance to the locations that would require compliance with 10 CFR 50.46, since the SMR-160 is not similar to a typical LWR reactor vessel. In evaluating the RAI response, the staff disagrees with Holtec's assertion that their unique design does not warrant special considerations and their assertion that there are no matters of special safety significance related to the combined vessel design and configuration. In this case, the safety significance is related to the potential consequence of failure of the locations, regardless of what standard the component is designed to meet.

The RAI response states, "there is no specific evidence that the emergency core cooling system (ECCS) would mitigate a rupture of the RPV in current pressurized water reactor (PWR), nor is there any regulation requiring it." This statement is unrelated to staff's RAI. The only reactor vessel failures that were historically excluded from typical light water reactors (LWR) were gross and serious failures of the vessel. The staff has not implied and does not intend the SMR-160 consider vessel ruptures, such as vessel wall failure or catastrophic head failures, as part of the design-basis. The staff's questions are focused on failures of the connections between the RPV/SG and SG riser. Consideration of an arbitrarily large break equivalent in size to reactor vessel inlet and outlet nozzles is well within the scope of 10 CFR 50.46 and GDC 35. As described in the RAI, these locations are subject to the LOCA regulations and an applicant to design or construct the SMR-160 must either demonstrate compliance or justify an exemption from the applicable regulations. For the purposes of the topical report, Holtec could establish design criteria that could serve as the basis for a future applicant's exemption request.

The RAI response provides no information on the acceptance criteria and design features that would mitigate a potential loss of coolant and does not provide design details that could serve as the basis for establishing criteria in support of a future exemption. Further, the RAI response does not provide any rationale for how 10 CFR Part 100 dose limits will be met for even minor breaks in the SG riser section.

The RAI response provides no information related to defense-in-depth or other operational considerations which would be credited, or additional design and analysis commitments, including criteria and assumptions that would be used to assess the consequences of failures that Holtec considers beyond-design-basis for the SMR-160 design. The RAI response and LTR should address what types and sizes of breaks are considered within the design-basis for the subject locations.

Design Details Related to Failure Prevention

The NRC requires detailed information regarding the welds associated with the Holtec planar inter-vessel forging (PIF) and SG riser design in order to evaluate whether the design criteria for these welds could form a sufficient basis for the staff to find that the probability of failure of these welds is sufficiently low and will not result in an undue risk to the public health and safety and is consistent with common defense and security. This finding is a necessary piece of any future exemption from 10 CFR 50.46. This information includes the materials, inspection, fabrication and resulting residual stresses, environmental and operating conditions, and their stress state.

The following are examples of the information required by the NRC to reach a conclusion that the failure probability of the welds is sufficiently low. After Holtec provides the information discussed below, the NRC may require additional specific information and commitments related to these issues to reach a conclusion regarding whether these welds have a sufficiently low probability of failure.

Most of the discussion in the RAI response is related to how the combined vessel (reactor) can operate safely without undue risk to the public. However, the issue for the topical report is whether Holtec can exclude consideration of LOCAs from the welds in the cross-duct (PIF) and SG riser, not whether the reactor meets the ASME BPV Code and can be operated safely. There is little detailed information or criteria provided in the RAI response or the topical report that can be used to form a basis to justify excluding consideration of LOCAs at these welds from the design basis and design basis accident analyses. Examples of information that could be used to support the LOCA exclusions, but for which the provided information is currently insufficient, are as follows:

- Information that Section XI of the ASME BPV Code will be used for preservice and inservice inspection provides no detailed criteria for inspections because the prescriptive nature of Section XI of the ASME BPV Code provides no inspections or definition for the type of the component/weld configuration such as PIF-to-SG weld and the SG riser welds. Holtec should specify and commit to the inspection method, frequency and acceptance criteria which will apply to the subject welds. Also, there is no objective evidence that 100% volumetric examination (by ultrasonic inspection) can be achieved for these welds. For example, the PIF-to-SG weld which is a corner butt weld with reinforcing fillet weld is similar to branch connections (weld-o-lets) which have experienced limited examination coverage. In addition, no information is available on the examination coverage and accessibility of the SG riser welds which was requested in the RAI. Holtec should provide specific details and commitments related to the above information.
- The PIF material elongation requirement imposed is lower and less conservative (X elongation) than the material specification for SA-508, Grade 3, class 2 (16% minimum elongation) and less than the material specifications and grades used in the operating reactor vessel fleet (18% minimum elongation). This is a measure of ductility, and the lower the number the less ductile the material is and more susceptible to degradation, such as brittle fracture or cracking. Since this is less conservative, Holtec should provide justification of the proposed material properties and how it will ensure sufficiently low probability of failure.
- Holtec should discuss why an XXX component (name not included since designated proprietary), which is a new component not discussed in the topical report previously, can prevent corrosion in the RCPB. Further, if this component is intended to be included in the design, Holtec should discuss the potential for failure of the component creating a crevice condition which could lead to degradation of the pressure boundary should be discussed.
- The RAI response does not provide sufficient basis to demonstrate that stress corrosion cracking is not an issue. Additional information should be provided by Holtec.
- No information was provided regarding how field welding of the cross duct (PIF) to the SG will be performed, and how the welding of this highly restrained weld might affect residual stresses (and what mitigation will be performed). In addition, no mention was made of how the SG riser and inner duct will be welded and any effects from these welds, particularly when welding to thick walled components such as the tubesheets. Holtec should provide specific details and commitments related to the above information.
- The RAI response does not provide the number of welds within the SG riser, nor the material specification for the riser. The staff also has additional concerns regarding the location of these welds since it might experience loadings from pressure differential, temperature differential, and environmental conditions (primary water on one side and secondary water on the other side of the riser). Holtec should provide specific details and commitments related to the above information including the pressure differential,

- temperature differential, and environmental conditions (primary water on one side and secondary water on the other side of the riser).
- For all welds associated with the PIF and SG riser, Holtec should provide the specific details and commitments regarding how they will be fabricated because the ASME BPV Code includes several methods and allowances/exemptions (for example post weld heat treatment exemptions) that may be necessary for these welds due to the configurations and highly restrained nature.
- The RAI response only states that the welds have low failure probability, similar to operating reactor vessels, and therefore should be excluded. However, for example, the staff notes that the material properties and inspection capabilities for the Holtec's combined vessel design discussed above are not similar to the current operating reactor vessels. Holtec should provide specific details and justification on the similarity and differences of the proposed excluded welds.
- The RAI response did not specify the methodology of the fracture mechanics analysis to be performed and the associated acceptance criteria to be met for future licensing activities. Holtec should describe deterministic and probabilistic fracture mechanic analyses (methodology and acceptance criteria) to demonstrate that the PIF and SG riser welds have a sufficiently low probability of failure.
- The RAI response did not address whether the PIF will be susceptible to thermal (aging) embrittlement or hydrogen embrittlement. Holtec should provide additional justification and commitments regarding why it is or is not susceptible to these degradation mechanisms.
- Holtec should identify all of the specific welds to be excluded from LOCA postulation, their configurations, weld joint details, dimensions and stress state including operational stresses in order to fully understand the basis for the Holtec position that these welds have sufficiently low probability of failure, and how they are similar/different than operating reactor vessel welds.
- The RAI response states that the combined vessel will be designed, fabricated, erected, and tested to the highest quality standards practical and monitoring systems will be employed to detect if there is any leakage from the RCPB. Holtec should clarify whether a local leakage detection system will be employed to detect any leakage from the RPV/SG connection and SG riser to support excluding the postulation of LOCAs at these components.
- The RAI response states that the structures within the SMR-160 are substantially better protected from flow induced vibrations than the current PWR operating fleet. Holtec should clarify whether the Holtec design will employ a comprehensive vibration assessment program consistent with the staff's guidelines as delineated in NRC Standard Review Plan (SRP) 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," and NRC Regulatory Guide (RG) 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing." If the comprehensive vibration assessment program will be submitted as part of a future licensing application, the NRC staff will review the acceptability of that comprehensive vibration assessment program at that point.
- The RAI response states that "If water or steam hammer loads are credible and significant, they will be included in the analysis. However, these loads are not expected to be the limiting loads on the welds or forgings of the Combined Vessel." Holtec should clarify whether this analysis will be performed as part of this topical report for evaluation in the current review, or as part of a future licensing action, at which point the NRC staff will conduct its detailed evaluation of water or steam hammer susceptibility and/or loading.