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

February 10, 2022

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 NOVEMBER 23, 2021, PUBLIC MEETING TO DISCUSS THE HOLTEC SMALL MODULAR REACTOR, SMR-160, TOPICAL REPORT: "ELIMINATION OF LARGE BREAK LOCA AND ESTABLISHMENT OF LOCA ACCEPTANCE CRITERIA"

On November 23, 2021, an Observation Public Meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and SMR, LLC, a Holtec International Company, regarding NRC staff's request for additional information regarding licensing topical report (LTR), "Elimination of Large Break Loss-of-Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria," Revision 2 (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 AccessionNo. ML21314A137 and was also posted on the NRC's public Web site.

This meeting discussed items related to the ongoing audit of the LTR and is a follow up to previous public meetings (June 16, 2021 (ADAMS Accession No. ML21180A465) and July 13, 2021 (ADAMS Accession No. ML21180A466)), to continue to formulate a path forward in determining if a potential break in the forged vessel-to-vessel connection between the reactor vessel and the steam generator can be considered a beyond design-basis accident and the basis for that determination. Enclosed are the meeting agenda (Enclosure 1), list of attendees (Enclosure 2), and the meeting summary (Enclosure 3).

Docket No. 99902049

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

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DATED: FEBRUARY 10, 2022

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ADAMS Accession Nos:
PKG: ML21364A118
MEMO: ML21364A119
MEETING NOTICE: ML21314A137 *** via e-mail** **NRR-106**

OFFICE	NRR/DNLR/NRLB: PM	NRR/DNRL/NRLB: LA	NRR/DNRL/NRLB: BC
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November 23, 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. – 2:30 p.m.	SMR-160 Licensing Topical Report Discussion – Open Session	NRC and Holtec
2:30 p.m.	Adjourn	

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November 23, 2021

List of Participants

Name	Affiliation
David Lewis	Holtec
Hickey, Kevin	Holtec
Lietwiler, Clay	Holtec
Marcille, Thomas	Holtec
McCloskey, Sean	Holtec
Morin, Tammy	Holtec
Rajkumar, Joseph	Holtec
Trotta, Rick	Holtec
Vehec, Jodine	Holtec
Daigle, David	Public
Ring, Mark	Numark Assoc
Seewald, Tom	Public
Brown, Christopher	NRC
Buford, Angie	NRC
Cranston, Greg	NRC
Dudek, Michael	NRC
Halnon, Gregory	NRC
Honcharik, John	NRC
Li, Yueh-Li	NRC
Manoly, Kamal	NRC
Mitchell, Matthew	NRC
Moyer, Carol	NRC
Nolan, Ryan	NRC
Patton, Rebecca	NRC
Scarbrough, Thomas	NRC
Tsao, John	NRC
Tseng, Ian	NRC
Barrett, Antonio	NRC

SUMMARY OF THE NOVEMBER 23, 2021, PUBLIC MEETING TO

DISCUSS THE HOLTEC SMALL MODULAR REACTOR, SMR-160, TOPICAL REPORT: "ELIMINATION OF LARGE BREAK LOCA AND ESTABLISHMENT OF LOCA ACCEPTANCE CRITERIA"

November 23, 2021

Meeting Summary

On November 23, 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), "Elimination of Large Break Loss-of-Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria," Revision 2 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML21068A255). The public meeting notice can be found in ADAMS under Accession No. ML21314A137 and was also posted on the NRC's public Web site.

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

The meeting discussed items related to the ongoing review and audit of the LTR and is a follow up to previous public meetings conducted on June 16, 2021 (ADAMS Accession No. ML21180A465) and July 13, 2021 (ADAMS Accession No. ML21180A466), to continue to formulate a path forward in determining if a potential break in the forged vessel-to-vessel connection between the reactor vessel and the steam generator (SG) can be considered a beyond design-basis accident (DBA) and provide the basis for that determination. Included in that determination is the applicability of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and the associated exemption request.

Holtec's position is that the postulation of a break of the SG to reactor pressure vessel (RPV) forged connection (planar inter vessel forging or PIF) and the SG riser should 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 (DBE). 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 and is welded to the tubesheets. The riser 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).

Holtec considers the PIF to be a vessel, rather than a pipe, and should therefore be treated as a Class 1 vessel in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code in a manner similar to the Class 1 RPV. Holtec stated that their PIF and a reactor vessel have the same operational

parameters and have the same external and internal operating pressures and temperatures. The PIF contains basically the same reactor coolant and boration as existing operating power plants. Therefore, Holtec believes that the PIF and RPV should have the same probability of failure and Holtec should not have to assume a break associated with the PIF since the Holtec plant and existing plants do not have to assume a reactor vessel break. A break in the Class 1 safety related RPV is considered a beyond design basis accident (DBA). Additionally, Holtec stated that they are taking steps to increase the conservatism on some of the ASME code calculations. Holtec also pointed out that the RPV gets embrittled from the neutrons and the PIF is not located in a high neutron area, which would reduce the risk of a break in a weld.

Therefore, Holtec is requesting in their LTR that postulation of a break of the SG to RPV forged connection (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 DBE.

NRC's position, as stated at a previous public meeting, is that 10 CFR 50.46, applies and that to not consider a break associated with the PIF as a DBE would require an exemption from 50.46. The NRC staff has stated that whether the PIF is called a pipe or a vessel, that 50.46 applies as a regulation and the basis for why an exemption is appropriate must still be addressed. Even though the exemption request would not be part of the LTR under review, the basis for an exemption and associated limitations and conditions regarding the PIF and associated welds, including tube sheet welds, needs to be included in the LTR.

NRC staff stated that if the LTR does not provide criteria or limitations and conditions that, if met after the design is finalized, would allow the NRC staff to make a determination that there is reasonable assurance that the PIF and associated welds can be exempt from 50.46, then the NRC staff will not be able to reach a finding in the LTR review that such a break would be a beyond DBA. And, consequently, Holtec would not gain any regulatory certainty regarding the acceptance of an exemption request in the future. At this stage, the design criteria or limitations and conditions that need to be met should be specified in the LTR by Holtec which would provide a framework to approve a future exemption request at a time when it could be demonstrated that the specified criteria or limitations and conditions could be met.

Holtec expressed concern that putting additional criteria, limitations and conditions in the LTR could bind a future applicant to meeting criteria that the applicant may not want. Holtec also indicated that they do not think that the LTR is the appropriate vehicle to stipulate and commit to certain guidance, such as NUREGs and regulatory guides, since they may not specifically apply when the design is finalized. Holtec stated that they are working on developing a list of the criteria that do apply and that need to be met to justify an exemption from 50.46, but that criteria and how the criteria will be met would be determined later by Holtec or by a future applicant. Holtec reiterated that they do not believe that the criteria can be determined now. And if criteria were developed now, it cannot be determined at this time how, or if, criteria stated in the LTR could be met later when an application is submitted since the design is not finalized. Therefore, Holtec does not want to make a commitment now that they don't necessarily agree with and may not be able to meet later.

NRC staff reiterated that in order for NRC to be able to make a finding that would provide some certainty that a future request for an exemption could be approved, additional criteria are needed. These additional criteria would be reviewed by NRC under 10 CFR 50.12, Exemptions, which requires that the staff find that the exemption would not cause undue risk to the public

health and safety. Additionally, the NRC will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever, for example: the exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the granting of the exemption; or compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated.

Therefore, the NRC staff has focused its review on risk and the two components that make up risk: probability and consequences. In parallel with the reactor systems staff review, the NRC mechanical and materials staff focused their review largely on aspect of the design that could reduce the probability of breaks at locations associated with the PIF.

NRC stated that the reactor system staff's analytical questions have focused on getting an understanding of the capability of the preliminary design to mitigate consequences for various break sizes associated with the PIF and associated welds, including more probable smaller breaks as well as defense in depth considerations. The preferred approach in conjunction with a LTR is for Holtec to specify the design criteria and for the staff to review it. If the Holtec specified design criteria are not sufficient to allow NRC staff to make a safety finding, then the NRC staff stated that they can provide criteria or limitations or conditions to be incorporated into the LTR to supplement existing criteria.

NRC staff stated that they have reviewed and written safety evaluation reports (SERs) on LTRs from applicants that identify criteria that must be met later, after an application has been submitted, to support future exemptions and after the design and associated analyses are finalized. The key is that the criteria to be met must be sufficiently comprehensive, such that when the criteria is later determined to be met, there is reasonable assurance that the plant will operate safely with no undue risk to the public health and safety. Therefore, through these discussions, the intent is to provide a path forward that gives Holtec reasonable certainty that when the criteria is determined to be met, there is a high likelihood that an exemption can be granted.

NRC staff stated that Holtec LTR provides a new and novel approach of using an acceptance criteria of only meeting the code as the sole acceptance criteria. Holtec stated that they felt that even though declaring the PIF a no break zone based on meeting the code is somewhat new and novel, there are other recent licensing topical reports for a similar topical area, which could be relied upon by NRC staff as precedent. Holtec expects that NRC would look to other similar applications and associated special requirements to see if a similar approach can be applied for this LTR. Holtec also stated that there is specificity and certainty in the requirements presented in their LTR that will be carried forward to an applicant with the understanding that Holtec is a designer and likely not the applicant or owner of the power plant. So, it is very critical that any special requirements, limitations and conditions provided by NRC set criteria clearly, with a high degree of specificity. And these requirements should be based on regulation and not on guidance. Holtec does not want to constrain the design requirements or place unnecessary commitments on a future applicant due to the NRC staff basing their LTR SER on those constraints. Holtec requested feedback on what requirements, limitations or conditions the NRC expects to add to the LTR to satisfy their regulatory and safety conclusions in the final SER.

NRC staff stated that it is difficult at this time to agree that the PIF is just like a typical reactor

vessel absent a more detailed design and analysis, which will not be available until the design is essentially final. Also, the design and configuration of the PIF is not the same as a typical pressure vessel (PV). The current discussion is qualitative, and it is difficult, absent a more detailed analysis, to come to the conclusion that a PIF break is a beyond DBE. An acceptance criterion in the topical report would require an analysis in the future to demonstrate the probability is extremely low and would provide the quantitative analysis to support the qualitative arguments that are currently laid out in the LTR and in ongoing meetings. Also, there may be results from a Holtec probabilistic risk assessment that supports the position that the probability of a break is very low such that it can be considered a beyond DBE.

Holtec stated that there is essentially no data regarding failure rates of a Class 1. And, because the ASME code must address different configurations, it is set up very conservatively such that by meeting the code there is an extremely low probability of failure.

NRC stated that with any application of a code the NRC will continue to lend all due credence to the ASME code and ASME code requirements. NRC understands that Holtec has extensive experience with application of the codes in the fabrication of large vessels. The concern is that from the nuclear construction standpoint the NRC has not seen fabrication and installation, including field welds, of such a forged fitting before. The forging is unique in shape and configuration and there is no operating experience. The reactor vessel, on the other hand is a right circular cylinder that has been installed at operating nuclear power plants for decades. NRC has some idea about the quality of construction and the RPV performance regarding maintaining integrity. Though the forging will be designed to basically the same rules, it is not fabricated in exactly the same way as a RPV. There will be differences in how the PIF is installed, welded, and inspected initially and periodically. In conjunction with the audit currently in progress, NRC staff is getting additional information to better understand the similarities and the differences between RPV fabrication and PIF fabrication and installation. NRC needs to better understand what is different that could potentially affect what the potential likelihood of failures may be and what would be the extent of that failure (break size). Since the design is not finalized, just saying the PIV will be fabricated and installed per the ASME code does not, as yet, provide sufficient assurance that the PIF meets a break exclusion criterion. To make a finding now in the SER for the LTR, additional limitations and conditions related to welds, materials used, inspection requirements, and other attributes, such as leakage monitoring, could provide a higher level of confidence that the PIF has a very low probability of failure and should a break occur, it would not exceed the break size analyzed as part of the design-basis. There would need to be a set of limitations and conditions to cover the key aspects of the PIF and associated welds. For example, there could be limitations and conditions on inspectability, welding, fatigue, and water chemistry. So, from both the reactor systems and the materials and mechanical points of view, Holtec needs to decide what would be the best set of potential limitations and conditions that, if met once the design is finalized, would provide sufficient information for the NRC to accept that a break of the PIF or associated welds is within the design-basis and that an exemption to 50.46 is justified.

NRC also stated that information is needed regarding the secondary side of the SG in conjunction with a potential primary to secondary reactor coolant leak. If the PIF weld to the SG tube sheet or riser should fail, there could be a primary to secondary side leak such that reactor coolant could enter the secondary side of the SG and bypass containment if not properly isolated by the main steam isolation valves. There is no discussion in the LTR regarding the secondary side isolation valves and the applicability of

10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” to prevent a release of radioactivity to the environment under those circumstances.

Although the audit confirmed additional information is necessary in the LTR, Holtec reiterated they would not provide any more information and requested the NRC staff proceed with developing limitations and conditions instead.

Regarding next steps, the NRC is continuing with the audit. Additionally, NRC staff is working on developing a set of potential limitations and conditions and is proposing that they could be discussed with Holtec to provide better insight on what could be added to the LTR to facilitate completing the review of the LTR and writing a SER. There would be several limitations and conditions that would be reviewed with Holtec at one or more public meetings.

Once sets of draft limitations and conditions are completed, public meetings will be scheduled. The development of the limitations and conditions will be coordinated between the reactor systems and the mechanical and materials groups and presented to Holtec as each set or group of sets is completed.

The meeting was adjourned.