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

January 31, 2022

MEMORANDUM TO: Michael I. Dudek, Chief
New Reactor Licensing Branch
Division of New Reactor Licensing
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: REGULATORY AUDIT REPORT OF THE HOLTEC SMR-160 TOPICAL REPORT HI-2201064 “ELIMINATION OF THE LARGE BREAK LOSS OF COOLANT ACCIDENT (LOCA) AND ESTABLISHMENT OF LOCA ACCEPTANCE CRITERIA,” REVISION 2

Enclosed is the U.S. Nuclear Regulatory Commission (NRC) staff’s Audit Report regarding the SMR, LLC (Holtec), Topical Report (TR) HI-2201064, “Elimination of the Large Break Loss of Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria.” The staff performed a teleconference audit of the Holtec TR with access to the Holtec electronic reading room. The purpose of the audit was to better understand the details of Holtec’s small modular reactor (SMR) design, as well as the bases for the assumptions and information provided in the TR.

By letter dated December 21, 2020 (ML20356A018), Holtec requested NRC review and approval of TR-HI-2201064. Holtec submitted a revision to TR-HI-2201064R2 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21064A037) on March 5, 2021, to provide additional information and criteria to describe the methodology regarding the regulatory basis to exclude LOCAs for any breaks associated with the vessel-to-steam generator (SG) connection and in the SG riser. The TR also provides a description of the passive core cooling system and the passive containment heat removal system. Additionally, the TR seeks to establish LOCA acceptance criteria and the basis for how these criteria are more restrictive than the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.”

During the review of the TR, the NRC staff issued three requests for additional information (RAIs): RAI 9832, issued on May 3, 2021 (ADAMS Accession No ML21123A187); RAI 9843, issued on May 20, 2021 (ADAMS Accession No. ML21140A440, closed); and RAI 9846, issued on August 6, 2021 (ADAMS Accession No. ML21190A238). RAI 9832 requested the justification supporting the sufficiency of the new proposed acceptance criteria by Holtec for the SMR-160 design, which was to include additional conservatism applied to the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), to provide reasonable assurance of adequate protection without addressing the applicability of regulatory requirements (such as 10 CFR 50.46). Responses to all three RAIs have been received; however, the staff determined that additional information was needed in order to close the RAIs. As such, the NRC staff initiated a regulatory audit to gain the additional needed information and the Audit Plan was issued on September 24, 2021 (ADAMS Accession No. ML21263A245). The audit was conducted over approximately 3 1/2 months, starting on October 4, 2021, and was completed on January 18, 2022. During the audit, the staff conducted periodic discussions with Holtec including possible responses to the RAIs. As the audit proceeded, the staff requested additional documents for review which were provided by Holtec in their electronic reading room (eRR).

To facilitate the closure of RAI 9832 and RAI 9846, the NRC staff conducted Observation Public Meetings on June 16, 2021 (ADAMS Accession No. ML21180A465) and July 13, 2021 (ADAMS Accession No. ML21202A235), between the NRC staff and Holtec. The purpose of these meetings was to discuss the Holtec response to RAI 9832 (ADAMS Accession No. ML21147A529) and to clarify the additional information needed for the NRC staff to accept the response and close RAI 9846. The meetings focused on obtaining more technical information regarding: (1) the forged connection between the SG and reactor pressure vessel (RPV), and (2) the SG riser; and to provide clarification on the additional technical information.

On November 23, 2021, an additional Observation Public Meeting (ADAMS Accession No. ML22028A023) was held between the NRC staff and Holtec, to discuss items related to the ongoing audit of the TR and as a follow-up to the previous public meetings on June 16, 2021, and July 13, 2021, to continue to formulate a path forward in determining if a potential break in the RPV to SG connection and in the SG riser can be considered a beyond design basis accident and the basis for that determination.

Per SMR, LLC, letter “Withdrawal of SMR, LLC, SMR-160 Topical Report: Elimination of the Large Break Loss of Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria (Project Number 99902049), dated January 18, 2022 (ADAMS Accession No. ML22018A171), Holtec has withdrawn the TR and plans to make the NRC staff aware of their approach going forward to use additional types of pre-applications engagements to gain perspectives on their SMR-160 design, such as through the use of White Papers, as they prepare for submitting documentation for a construction permit application.

As a result of Holtec’s withdrawal letter, the staff has subsequently concluded the audit, has stopped the review of the subject TR, and will be closing the associated Cost Activity Code regarding fee-billing.

Docket No. 99902049

Enclosure: As stated

cc w/encl: Listserv

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SUBJECT: REGULATORY AUDIT REPORT OF THE HOLTEC SMR-160 TOPICAL REPORT HI-2201064 “ELIMINATION OF THE LARGE BREAK LOSS OF COOLANT ACCIDENT (LOCA) AND ESTABLISHMENT OF LOCA ACCEPTANCE CRITERIA,” REVISION 2 DATED: JANUARY 31, 2022

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REGULATORY AUDIT REPORT OF THE HOLTEC SMR-160 TOPICAL REPORT HI-2201064 "ELIMINATION OF THE LARGE BREAK LOSS OF COOLANT ACCIDENT (LOCA) AND ESTABLISHMENT OF LOCA ACCEPTANCE CRITERIA," REVISION 2

Applicant:	SMR, LLC., A Holtec International Company
Applicant Contact:	Tammy Morin
Date:	October 4, 2021 – January 18, 2022
Location:	U.S. Nuclear Regulatory Commission (NRC) Headquarters/ Teleconferencing (via Holtec International, LLC Electronic Reading Room (eRR))
Audit Team:	Antonio Barrett, Office of Nuclear Reactor Regulation (NRR) Greg Cranston (NRR) John Honcharik (NRR) Renee Li (NRR) Ryan Nolan (NRR) Thomas Scarbrough (NRR) Rosemary Sugrue (NRR) Zach Sweeney (NRR) John Tsao (NRR)
Holtec Staff Participants:	Kevin Hickey, Manager, Mechanical Design, SMR Jodine Jansen Vehec, Manager, Probabilistic Safety, SMR Clay Lietwiler, Manager, Deterministic Safety, SMR Sean McCloskey, Probabilistic Safety Analysis Engineer, SMR Tammy Morin, Sr. Licensing Manager Joseph Rajkumar, Manager, System Engineering, SMR Rick Trotta, Director, SMR-160 Development Program, SMR

Audit Scope and Purpose

The scope of this audit included reviewing supporting design documents, design drawings, calculations, methodology, and other related information supporting Topical Report (TR) HI-2201064, "Elimination of the Large Break Loss of Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria," (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21064A038) and meeting with subject matter experts to discuss details of the information.

The purpose of the audit is to better understand the details, bases, and assumptions for the information provided in the TR used to justify Holtec's approach to eliminate the need to analyze for LOCAs at the RPV to SG connection and SG riser locations and associated regulatory implications.

Additionally, the audit was conducted to facilitate the closure of RAI 9832 (ADAMS Accession No. ML21147A529) and RAI 9846 (ADAMS Accession No. ML21190A238) and as a follow-up to previous meeting discussions. The NRC staff had previously conducted Observation Public

Meetings on June 16, 2021 (ADAMS Accession No. ML21180A465) and July 13, 2021 (ADAMS Accession No. ML21202A235), between the NRC staff and Holtec. The purpose of those meetings was to discuss the Holtec response to RAI 9832 and to clarify the additional information needed for the NRC staff to accept the response and close RAI 9846. Those meetings focused on obtaining more technical information regarding: (1) the forged connection between the steam generator (SG) and reactor pressure vessel (RPV), and (2) the SG riser; and to provide clarification on the additional technical information needed.

A list of Holtec documents available for this audit is included in Table 1, List of Documents for the SMR-160 Audit on “Elimination of the Large Break Loss of Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria,” which was provided by Holtec during the audit.

Table 1 List of Documents for the SMR-160 Audit on “Elimination of the Large Break Loss of Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria”

File Index	Audit Document File	Description
4.1	09Nov21 - Holtec responses - EMIB Proposed Update to Holtec TR (R1).docx	EMIB Proposed Update Items to Holtec Licensing Topical Report (LTR) HI-220-1064R1, “Elimination of Large Break Loss of Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria”
4.2	Draft HI-2201064R3-redline.pdf	Elimination of Large Break Loss of Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria for SMR-160 Report No. HI-220-1064 Rev 3
4.3	Mapping Audit Plan to Draft Response.xlsx	Mapping Audit Plan to Draft Response
4.4	Response to 9832 meeting minute bullets - responses.docx	Response to 9832 meeting minute bullets - responses
4.1.1	Accumulator.pdf	Drawing of accumulator
4.1.2	CISO info for MSS and MFW.docx	Containment penetrations and isolation valves on main steam and main feedwater systems – Table 8-1
4.1.3	Containment heights.pdf	Drawing of inside containment plan and elevation views
4.1.4	DSA Responses NRC Audit Call 11-9-2021.docx	DSA Responses NRC Audit Call 11-9-2021 re: containment flood-up volume vs height, containment condensation rate
4.1.5	DVI LOCA.docx	Document describing plant response to direct vessel injection double ended guillotine break LOCA
4.1.6	Inner diameters.pdf	Reactor pressure vessel and shroud and steam generator internals drawing

4.1.7	PCHR PID DWG-11256-R2.0.pdf	Passive containment heat removal system drawings
4.1.8	PCMWS PID DWG-10999-R2.0.pdf	Passive core makeup water system piping & instrumentation Diagram
4.1.9	PDHR PID DWG-10963-R2.0.pdf	Pressure decay heat removal system and instrumentation diagram
4.1.10	RCS PID DWG-10960-R0.0.PDF	Reactor coolant system P&ID
4.1.11	RCS, SG, Containment volumes.pdf	Table showing RCS, SG and containment volumes
4.1.12	Responses to some NRC Questions from Nov 4.docx	Responses to some NRC Questions from Nov 4
4.1.13	RT and ESF Logic diagram DWG-11156-R1.0.pdf	Reactor trip and emergency safeguard features actuation logic diagram
4.1.14	Safety Classification Matrix from HPP-160-3004-R5.pdf	Systems, structures and components classification standard for SMR-160 table
4.1.15	SDHR PID DWG-10994-R2.0.pdf	Secondary decay heat removal system P&ID
4.2.1	CVCS PID DWG-10957-R2.0.pdf	Chemical and volume control system P&ID
4.2.2	Main Feedwater System PID DWG-10966-R2.0.pdf	Main feedwater system P&ID
4.2.3	Main Steam System PID DWG-10967-R1.0.pdf	Main feedwater system P&ID
4.2.4	RHR PID DWG-10964-R2.0.pdf	Residual heat removal system P&ID
4.2.5	Spent Fuel Pool Cooling System DWG-10965-R1.0.pdf	Spent Fuel Pool Cooling System P&ID
4.3.1	Basis for size SG Riser break .docx	Basis for size SG Riser break document
4.3.2	LOCA Case Descriptions.docx	LOCA cases document
4.3.3	SMR-10_R-LMC-T0104r0.pdf	RELAP accident analysis form
4.3.4	SMR-10_R-LMC-T0105r0.pdf	RELAP accident analysis form
4.3.5	SMR-10_R-LRU-T0353r0.pdf	RELAP accident analysis form
4.3.6	SMR-10_R-LRU-T0354r0.pdf	RELAP accident analysis form
4.4.1	Changes to be incorporated into TR - 12Nov21 welds call.docx	Changes to be incorporated into Topical Report related to ASME Code, inspections, and welding
4.4.2	Minimum distances.pdf	Associated with RPV, PIF, and SG

4.4.3	Pressurizer.PDF	Pressurizer general arrangement drawing
4.4.4	Question for John T and John H.docx	Questions related to welding
4.4.5	None	None
4.4.6	RCS weld locations - 12Nov21.pdf	Drawing showing vessel to vessel connection welds
4.4.7	Reactor Assembly.pdf	Reactor assembly drawing
4.4.8	Reactor Coolant System.pdf	RCS interface general arrangement drawing
4.4.9	Reactor Pressure Vessel.pdf	RPV general arrangement drawing
4.4.10	Reactor Pressure Vessel Shell.pdf	Reactor Pressure Vessel Shell drawing
4.4.11	Steam Generator GA.pdf	Steam generator general arrangement drawing
4.4.12	Steam Generator Lower Head.pdf	Steam Generator Lower Head drawing
4.4.13	Steam Generator.pdf	Steam generator internals drawings
4.4.14	Steam Generator Tubesheets.pdf	Steam generator tubesheet drawings
4.4.15	Weld and examination geometry.pdf	Weld and examination geometry drawings

Audit Summary

During a pre-audit meeting on September 27, 2021, the NRC staff made introductory remarks regarding the regulatory audit background, scope, objectives, and agenda. This included the regulatory basis for the staff's audit and the office instruction for conducting a regulatory audit. The staff stated that the purpose of this regulatory audit was to better understand the details, bases, and assumptions for the information provided in the Holtec TR. The audit commenced on October 4, 2021.

The staff audited information related to multiple aspects of the SMR-160 design. In addition, the staff held multiple audit discussions with Holtec via telephone conferences. The Holtec electronic reading room (eRR) contained information previously developed and information added that supported staff questions and draft request for additional information (RAI) responses.

Per SMR, LLC, letter, "Withdrawal of SMR, LLC, SMR-160 Topical Report: Elimination of the Large Break Loss of Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria (Project Number 99902049), dated January 18, 2022 (ADAMS Accession No. ML22018A171), Holtec has withdrawn the TR and plans to make the NRC staff aware of their approach going forward to use additional types of pre-applications engagements to gain perspectives on their SMR-160 design, such as through the use of White Papers, as they prepare for submitting documentation for a construction permit application.

Summaries relevant to the design aspects audited are provided below:

Mitigation Capability and Defense-In-Depth:

During the audit, the NRC staff focused on the design mitigation capability and defense-in-depth for potential breaks in the reactor pressure vessel and steam generator (RPV/SG) connection and in the SG riser (also referred to as “subject locations”).

As part of the audit, the NRC staff discussed with Holtec mitigation capability, defense-in-depth or other operational considerations which might be credited, or additional design and analyses, including criteria and assumptions, that could be used to assess the consequences of breaks. The NRC staff reviewed the audit material in the eRR as follows:

1. Drawings and documents providing the key volumes for the reactor coolant pressure boundary (RCPB) and containment. Engineering calculation documents for analyses performed for inventory loss from the RCPB.
2. P&ID drawings for the RCPB and connected systems.
3. Drawings and documents providing the dimensions of key components in the RCPB.
4. Drawings and documents providing the safety classification for the associated valves and piping for the RCPB and connecting systems including secondary side.
5. Documents containing design information for the heat removal systems and containment.
6. Documents containing system actuation logic diagrams, isolation signals and requirements.

Mechanical engineering-related topics for potential mechanical failure and degradation mechanisms

During the audit, the NRC staff had multiple discussions with Holtec regarding certain mechanical engineering-related information related to potential mechanical failure and degradation mechanisms with respect to the RPV/SG connection and in the SG riser. As part of the audit, the NRC staff discussed issues with Holtec related to potential degradation mechanisms of concern including leakage detection, vibration, and water/steam hammer. The NRC staff reviewed the audit material in the eRR as follows:

1. TR markups related to potential mechanical failure and degradation mechanisms.
2. Responses to audit items related to potential mechanical failure and degradation mechanisms.
3. Drawings and documents providing details related to the RPV/SG connection and SG riser.

Materials engineering-related topics for the potential for leakage and/or structural failure

During the audit, the NRC staff focused on the determination of extremely low probability of failure of (1) the PIF (planar inter vessel forging; aka connection duct) to SG welds, (2) the SG riser section welds, and (3) the SG riser to tubesheet welds.

As part of the audit, the NRC staff discussed with Holtec details regarding materials, installation, inspections, residual stress analysis and fracture mechanics analysis on November 12, 2021. The NRC staff reviewed the audit material in the eRR as follows:

1. Several arrangement drawings in the 4.4 Materials section in the eRR for the RPV and SG were reviewed to understand the locations of the subject welds, interferences with other connections/obstructions, and the accessibility to these welds for the required inspections, including construction, preservice and inservice inspections.
2. Arrangement and detail drawings for the RPV, SG, and SG tubesheets were reviewed to gather specific design information of the components and weld design configuration, and how adjacent components or internal structures interact with the subject welds. The NRC staff also reviewed these drawings to gather specific design information of the subject welds and how the environment at each location could either enhance or mitigate material degradation.
3. Proposed responses to meeting minute bullets in the eRR were reviewed to address welding, including base material, filler metal, base material processing, specific welding sequencing regarding residual stresses and other processes that could cause degradation such as hydrogen embrittlement. Proposed responses to meeting minute bullets in the eRR were reviewed concerning weld residual stress and fracture mechanics analysis, including methodology and acceptance criteria.
4. Detail drawings for the RPV and SG, and the proposed responses to meeting minute bullets in the eRR were reviewed to understand how base material, filler metal and the material processing could affect the material properties and increase susceptibility to material degradation.

<u>Open Items/Identified Gaps:</u>

Per SMR, LLC, letter, "Withdrawal of SMR, LLC, SMR-160 Topical Report: Elimination of the Large Break Loss of Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria (Project Number 99902049), dated January 18, 2022 (ADAMS Accession No. ML22018A171), Holtec has withdrawn the TR. However, as part of the NRC staff's review, the following open items/identified gaps were identified.

1. In the area of mitigation capability and defense in depth, the following items should be considered:

    a. A methodology to determine the appropriate range of breaks at the subject locations for 10 CFR 50.46 design basis analysis break exclusion based on break probability considering a full spectrum of breaks. Breaks at the subject locations that do not meet the methodology threshold for exclusion from 10 CFR 50.46 analyses based on break probability should be included in a discussion section similar to TR Section 3.5, "Identification of Potential Break Sizes and Locations", and associated subsections. Discuss how the initiating event frequency, based on break probability, will be determined (e.g., analytical calculation or evaluation) at the subject locations. Additionally, the methodology should address uncertainties and in particular uncertainties associated with novel design features.
    b. Analysis commitments to perform analyses to address the SMR-160 design's capability to mitigate coolant inventory releases for breaks that are larger than those that are excluded from 10 CFR 50.46 design basis analyses at the subject locations or an alternate design approach that includes design attribute criteria that mitigate coolant inventory releases at the subject locations. Detail the defense-in-depth capabilities of the design and include criteria and assumptions that would be used to assess the consequence of failures at the subject locations.

c. In order to use the probabilistic risk assessment (PRA) protocol to support defense-in-depth aspects of the SMR-160 design, include a commitment or detailed discussion for how the initiating event frequency, based on break probability, for a spectrum of breaks up to and including a double ended guillotine break at the subject locations will be applied. The commitment or discussion should include how thermal hydraulic analyses for the break spectrums will be performed to support the core damage frequency calculations in the PRA.
d. Given the unique configuration of the Holtec SMR-160 design, a postulated break in the SG riser could result in the release of significant quantities of radioactive fission products due to potential failure of all three fission product barriers. Discuss the defense-in-depth approach regarding the three barriers for the release of radioactive material to the environment or an alternative approach for a postulated break in the SG riser. Include a commitment or discussion of how any components connected to the SG and penetrating containment shall provide leak detection, isolation and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance of safety of isolating these systems. Additional discussion of any valves relied on to isolate a breach of the reactor coolant system (RCS) is included with respect to appropriate 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” leakage rate testing. Discussion of the design’s capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits should be included.
e. Include a discussion of the applicable containment and dose requirements with respect to the portions of the design that are part of the RCPB, which includes the subject locations, and describe how they remain inside primary containment or an alternate approach. The discussion of related regulations for the RCPB for these locations should include 10 CFR 50.34(a)(1)(ii) with respect to allowable radiological dose consequences for postulated accidents and GDCs 50 through 57 with respect to containment requirements.

2. After reviewing the above information in the eRR, the NRC staff determined that the current design information in the TR would need additional information regarding materials engineering-related topics which would have a significant impact on the potential for leakage and/or structural failure at the subject locations. This additional information would be necessary to support finding that there is high confidence in the extremely low likelihood of failure at the subject locations. The staff has communicated in public meetings, RAIs, audit sessions, etc., specific materials engineering-related information which should be submitted. For this area, the following items should be considered:
   a. Materials of construction (both base material and weld material, including their material properties, material type, material processing, etc.)
   a. Environment (e.g., water chemistry controls and monitoring; operating temperatures; presence of crevice conditions, stagnant coolant, large heat sinks affecting residual stresses, etc.)
   b. Elements of weld design and weld geometries
   c. Fabrication, including welding method, weld residual stresses and analysis, controls on welding, welder qualification, post weld heat treatment, preheat etc.)

d. Nondestructive examination for fabrication, preservice and inservice (including accessibility for inspection to achieve 100% volumetric examination coverage of weld and adjacent base material, methods of inspection, acceptance criteria, qualification, etc.)
e. Supporting fracture mechanics analyses with methodology and acceptance criteria demonstrating an extremely low likelihood of leakage and/or structural failure of the PIF (planar inter vessel forging; aka connection duct) to SG welds, the SG riser section welds, and the SG riser to tubesheet welds.

3. During the audit, the NRC staff had discussions with Holtec regarding certain mechanical engineering-related information. The staff communicated that the TR does not adequately document how a future applicant will address potential mechanical failure and degradation mechanisms to support elimination of breaks as proposed in the TR. The specific mechanical engineering-related information which should be considered include the following:
   a. The NRC staff determined that the TR did not provide an adequate assessment of the potential failure and degradation mechanisms resulting from vibration conditions, or a description of a comprehensive vibration assessment program (CVAP), to provide reasonable assurance that the potential for a break at the RPV/SG connection or in the SG riser is extremely small to support a 10 CFR 50.12 exemption request. NRC Standard Review Plan (SRP) Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," and Regulatory Guide (RG) 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Startup Testing," provide guidance for assessing degradation mechanisms from vibration conditions and for describing an acceptable CVAP for a new reactor. The NRC staff will need to review a future applicant's evaluation of the potential failure and degradation mechanisms from vibration conditions, and establishment of an adequate CVAP, to support a 10 CFR 50.12 exemption request for an SMR-160 reactor.
   b. The NRC staff determined that the TR did not provide an adequate assessment of the potential failure and degradation mechanisms resulting from steam/water hammer conditions, or a description of the design features and operational procedures that will preclude steam/water hammer conditions or accommodate the potential steam/water hammer loads, to provide reasonable assurance that the potential for a break at the RPV/SG connection or in the SG riser is extremely small to support a 10 CFR 50.12 exemption request. The NRC staff will need to review a future applicant's evaluation of the potential failure and degradation mechanisms from steam/water hammer conditions for consideration of a 10 CFR 50.12 exemption request for an SMR-160 reactor.
   c. The NRC staff determined that the TR did not provide an adequate description of a leakage detection and monitoring system that will be employed to detect any leakage at the welded locations in the RPV/SG connection and SG riser, to support a 10 CFR 50.12 exemption request. In addition, the TR did not include proposed Technical Specifications for action to be taken if any leakage from the RPV/SG connection or SG riser in the SMR-160 reactor is identified such that the reactor will be shut down promptly to reduce the RCS pressure and minimize the potential for a large break at those locations, that satisfy the 10 CFR 50.36 requirements. The NRC staff considers potential SG riser leakage to be similar to RCPB leakage because of the potential for offsite radioactive release. RG 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," provides guidance for developing a leakage detection and monitoring

system for a new reactor. The NRC staff will need to review a future applicant's plans for an effective leakage detection and monitoring system (including instrument sensitivity and locations), and proposed Technical Specifications, to provide reasonable assurance of immediate identification of any leakage from the RPV/SG connection or in the SG riser with appropriate action requirements, to support a 10 CFR 50.12 exemption request for an SMR-160 reactor.