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Recipients:
"Holtec-SMR-160RAIsPEm Resource" <Holtec-SMR-160RAIsPEm.Resource@nrc.gov>
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QUESTIONS

NONE-1

Request for Additional Information- Holtec Topical Report HI-2201064, Revision 2, "Elimination of the Large Break Loss of Coolant Accident (LOCA) and Establishment of LOCA Acceptance Criteria," for SMR-160 Small Modular Reactor

Regulatory Requirements

The Introduction of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 states, in part, that an applicant must consider the design of a specific facility and satisfy the necessary safety requirements with respect to type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary (RCPB) in determining design requirements to suitably protect against postulated loss-of-coolant accidents. Further it defines loss of coolant accidents (LOCAs) as “those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system (RCS).”

General Design Criterion (GDC) 35, “Emergency Core Cooling,” in 10 CFR Part 50, Appendix A, requires, that a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. In addition, suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The NRC regulations in 10 CFR 50.46 require, in part, that emergency core cooling system (ECCS) cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. The NRC regulations in 10 CFR 50.46 define LOCAs as hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS. The requirements of 10 CFR 50.46 are in addition to any other requirements applicable to the ECCS set forth in 10 CFR Part 50. The criteria set forth in 10 CFR 50.46(b), with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in 10 CFR Part 50, including in particular GDC 35 of 10 CFR Part 50, Appendix A.

The NRC regulations in 10 CFR 100.21, "Non-seismic siting criteria," refer to 10 CFR 50.34(a)(1) with respect to allowable radiological dose consequences for postulated accidents. The radiological dose criteria for the exclusion area boundary and low population zone dose consequences is set at 25 rem total effective dose equivalent (TEDE) for events of extremely low probability. With respect to the content of the application, 10CFR 50.34(a)(1)(ii) states that it is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products.

The above regulations are applicable to the reactor pressure vessel (RPV) to steam generator (SG) connection, and SG riser locations (also referred to below as "subject locations") in the Holtec SMR-160 small modular reactor and need to be addressed irrespective of how the locations are categorized.

Discussion

Section 3.2, "Design Requirements," of Topical Report HI-2201064 (Revision 2) lists design requirements for the Combined Vessel of the Holtec SMR-160. The design requirements listed in Section 3.2 focus on the design of the Combined Vessel in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, and the inservice inspection (ISI) requirements in the ASME BPV Code, Section XI.

In its topical report, Holtec is requesting that the postulation of breaks at specific locations be excluded from the SMR-160 design basis based on its assumption that a break in the subject locations have a very low probability of occurrence. Therefore, breaks at the subject locations are not considered by Holtec to be within the scope of the NRC regulations listed above. Based on the information provided by Holtec to date, the NRC staff cannot reach the same conclusion. The staff considers the components for the outlet of the RPV and SG riser to function as a pipe; and therefore, failures of these components must be considered within the licensing basis of the facility and are subject to the regulations identified above. In addition, the RPV/SG connection is sufficiently unique such that it warrants special consideration with respect to the treatment of this configuration. The primary function of the external connection between RPV outer inlet nozzle and SG is to act as a fluid coupling to transfer water between the SG to the RPV; and therefore, it is more appropriate for the RPV/SG connection to be treated as a pipe within the context of 10 CFR 50.46[1].

Generally, RCPBs for nuclear power plants are required to be designed and constructed in accordance with the ASME BPV Code to meet certain NRC regulations. Dynamic effects of postulated pipe breaks may be excluded from the design basis if analyses reviewed and approved by the NRC demonstrate that the probability of fluid system piping rupture is "extremely low" under conditions consistent with the design basis for the piping. In order to exclude the dynamic effects, the design basis must meet the requirements of the ASME BPV Code as incorporated by reference in 10 CFR 50.55a, and additional considerations in order to meet the “extremely low” requirement which is beyond meeting the ASME BPV Code. Nonetheless, satisfying minimum ASME BPV Code or exclusion of dynamic effects requirements still requires the postulation of breaks in the RCPB within the safety analysis. Absent the presence of special circumstances, as defined in 10 CFR 50.12, failure of these connections should be analyzed as design-basis events unless an exemption to requirements identified above is justified. Special circumstances may be justified to exclude postulating a LOCA at the specified locations from the design basis if Holtec can provide appropriate justification and conservative acceptance criteria that demonstrates a sufficiently low likelihood of failure at the subject locations with acceptably low consequences. Excluding postulating a break from the design basis and/or design basis accident analyses entirely would, in part, require meeting the minimum necessary for exclusion of the dynamic effects and additional considerations that demonstrate that the probability of fluid system piping rupture is sufficiently low (i.e., beyond “extremely low”) to determine that reasonable assurance of adequate protection is provided. As such, any proposed topical report acceptance criteria must allow the NRC staff to have a similar level of reasonable assurance of adequate protection that has been provided by the RCPB and ECCS designs that have been previously reviewed and approved for GDC 35, 10 CFR 50.46 and 10 CFR 100.21 considerations.

In order to make a decision to exclude certain break locations from the Holtec design, a sufficient level of confidence is needed so that the likelihood of a break at either of these locations will remain extremely low for the life of a Holtec reactor. To obtain such a level of confidence, the fundamentals of how and why a particular location in a design might be subject to failing during service should be considered. The potential for failure can generally be related to: (1) the likelihood for some mode of degradation to occur at the location and to develop to a sufficient extent that, (2) the location becomes prone to fail during some design basis loading event (e.g., an SSE).

The propensity for degradation to occur at any location can usually be related to three fundamental attributes: (1) the material of construction (which includes how the component or weld was fabricated), (2) the environment in which the location exists (e.g., water chemistry, temperature, radiation, etc.), and (3) the stresses the location is subject to during normal operation (which may include pressure, thermal, weld residual, etc.). Improving upon or conservatively limiting these attributes holistically to the extent practical will lead to a higher level of confidence that modes of degradation will not occur, or if degradation does occur, the expectation that it will progress less quickly providing opportunity to identify the degradation via inservice inspection (ISI), or possibly leakage detection, and take action to prevent it from progressing to the point where it could lead to failure during a design basis event.

<u>Reactor Pressure Vessel to Steam Generator Connection</u>

The RPV to SG connection identified in Section 3.3 of the topical report is described as being an integral part of the Combined Vessel. The Combined Vessel is part of the RCPB and is subject to GDC 35 and 10 CFR 50.46 requirements. It is recognized that Holtec has designated the RPV to SG connection as part of the vessel. Regardless of the location’s designation[2], the NRC staff needs additional information to find that the Combined Vessel meets the high degree of reliability needed to provide reasonable assurance of adequate protection. Traditional reactor vessels have been constructed and operated well within their design limits and, therefore, serious breaks or ruptures in the vessel have been considered to be much less likely to occur than a break in other portions of the RCPB[3]. In addition, the longstanding practice for designing the ECCS for the largest pipe break in the RCS results in the design having an ECCS capability and capacity that can well exceed most failure mechanisms of a reactor vessel; thus, providing robust defense-in-depth for operating reactors. It is unclear to the staff what defense-in-depth is provided for the SMR-160, and whether ECCS could perform its safety function if the RPV/SG welded connection were to fail.

<u>Steam Generator Riser</u>
The SG riser identified in Section 3.4 of the topical report is described as being an integral part of the SG. The SG riser is part of the RCPB and is subject to applicable portions of 10 CFR 100.21, as well as GDC 35 and 10 CFR 50.46 requirements. It is recognized that Holtec has designated the SG riser as an integral part of the SG. Regardless of the location's designation[4], the staff needs additional information to find the design can meet identified regulatory requirements. The SG riser forms the boundary of the SG and, as such, a rupture at this location can cause a large inventory release that bypasses the containment that is significantly larger than the single SG tube rupture assumed for design basis accident analyses. Multiple barriers for the release of significant quantities of radioactive fission products should exist. The three primary barriers are: (1) fuel rod cladding, (2) reactor coolant system pressure boundary, and (3) containment. During postulated accidents, failure of one or two of these primary barriers is acceptable, provided radiological dose consequences are within allowable limits. The allowable consequence for a postulated accident is based on the probability (i.e., expected frequency) for that accident and may be set as less than criteria established for events of extremely low probability.

Given the unique configuration of the Holtec SMR-160 design, a postulated break in the SG riser could result in failure of all three primary barriers for the release of significant quantities of radioactive fission products. The initiating event, failure of the SG riser, removes the reactor coolant system pressure boundary. The sudden depressurization and loss of coolant inventory would challenge the integrity of the fuel rod cladding and primary inventory releases directly to the secondary side, bypassing containment. To support a risk-informed decision of Holtec's unique configuration, the staff needs a better understanding of both: (1) the probability of SG riser failure, and (2) the consequence of such a failure.

Requested Additional Information

The NRC staff requests Holtec to provide justification for the SMR-160 design to support that the proposed acceptance criteria, with the additional conservatism applied to the ASME BPV Code, are sufficient, and whether additional acceptance criteria are needed to provide reasonable assurance of adequate protection. The acceptance criteria and methodology used in an analysis to eliminate a LOCA at the RPV/SG connection and SG riser in the Holtec SMR-160 design should consider addressing the following:

1. the potential for unusual operating conditions which could contribute to degradation or failure (for example, thermal stratification, unexpected vibration, water hammer, etc.) or any degradation mechanism, including environmental conditions (thermal, pressure, water chemistry, stresses, etc.) of each weld location that could compromise its structural integrity;
2. experience with similar combined vessel construction and performance in the nuclear industry and other industries;

3. material and construction methods including welding, heat treatment, field vs. shop welding, fit-up and weld residual stresses, nondestructive examination, welding joint configurations and dimensions, etc.;
4. details of the configuration of components and welds and how they interact with each other;
5. fracture mechanics analyses (deterministic or probabilistic) and acceptance criteria for the welds which would demonstrate the hypothetical margin to failure should a flaw exist during service;
6. accessibility for ASME BPV Code, Sections III and XI preservice inspections (PSI) and ISI examinations and additional methods/frequency of ISI or possibly leakage detection that could improve the likelihood that, should a mode of degradation develop, it will be identified and addressed as early as possible;
7. additional limits on the stresses the locations will see during normal operation (for example, reducing the allowable stress limits for the design due to pressure, thermal, etc., relieving weld residual stresses, etc.) and during the limiting design basis loading condition to minimize the likelihood of failure if degradation were to occur and before it is detected and addressed;
8. operational attributes, processes, and defense-in-depth 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 are considered beyond-design-basis for the Holtec SMR design; and
9. the consequences of failure for the SG riser location specifically. The discussion should provide a detailed sequence of events of the accident progression for a SG riser failure for understanding of the consequences, including the response of safety-related systems, structures, and components and any credited operator actions to mitigate the consequences (e.g., core damage, and release of fission products to the environment), recover and restore core inventory, and remove long-term decay heat. Information concerning the capacity and capabilities of primary and secondary heat removal systems should be included. If applicable, discuss the capability of secondary side steam piping to withstand the weight of liquid inventory.

The topical report should be updated accordingly to address the items above and align with applicable regulations. For example, if compliance with pertinent regulatory requirements is not intended to be demonstrated, the topical report should identify that a future applicant referencing this topical report is required to submit an exemption request to the applicable regulation(s).

[1] This viewpoint is consistent with Paragraph U-1(c)(2) of ASME BPV Code, Section VIII-1, which defines piping systems as those “structures whose primary function is the transport of fluids from one location to another within a system of which it is an integral part” while vessels are defined as “containers for the containment of pressure, either internal or external.” This definition is also consistent with the Companion Guide to the ASME BPV Code (available from ASME), and the previously communicated staff interpretation on this topic.

[2] As noted above, the NRC staff does not consider this component to be a vessel. Even if the staff could reach a determination that the location can be treated as a vessel, the staff would still need to consider potential failures of this configuration as part of the licensing process due to the uniqueness of this design and its safety significance. Historically, the staff has determined that traditional reactor vessels have a low risk of catastrophic failure and would not rupture in a manner which would prevent ECCS from performing its safety function. The Commission has stated that protection against pressure vessel failures is subject to the licensing process if there are special considerations present and special safety significance. This is further detailed in "Issuances of the Atomic Energy Commission," Volume 7, Atomic Energy Commission Reports, Opinions and Decisions, ALAB-188, pp. 326-328.

[3] See ECCS Hearing Transcript, RM-50-1, Volume 25, April 11, 1972.

[4] See footnote 2.