NuScaleDCRaisPEm Resource

From: Cranston, Gregory
Sent: Friday, February 1, 2019 7:28 AM
To: Request for Additional Information
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Subject: Request for Additional Information No. 516 eRAI No. 9647 (15.0.6)
Attachments: Request for Additional Information No. 516 (eRAI No. 9647).pdf

Attached please find NRC staff's request for additional information (RAI) concerning review of the NuScale Design Certification Application.

Please submit your technically correct and complete response by March 3, 2019, to the RAI to the NRC Document Control Desk.

If you have any questions, please contact me.

Thank you.

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Request for Additional Information No. 516 (eRAI No. 9647)

Issue Date: 01/31/2019
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 15 - Introduction - Transient and Accident Analyses
Application Section: 15.0.6

QUESTIONS

15-29

The initiating event of the rod ejection accident is the rapid ejection of a control rod (CR) caused by an assumed control rod housing failure as described in Standard Review Plan (SRP) Section 15.4.8 and Regulatory Guide 1.77. The requirement to evaluate the rod ejection accident is given by General Design Criterion (GDC) 28 which states,

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Consistent with GDC 28, SRP 15.0 classifies control rod ejection as a postulated accident as does NuScale in Final Safety Analysis Report (FSAR) Section 15.4.8.1. A conservative initial condition assumes the regulating control rods are at rod insertion limit defined by technical specifications. Following the rod ejection, a reactor trip occurs on either a high flux rate, high flux or high pressurizer pressure inserting the remaining control rods. The emergency core cooling systems (ECCS) will actuate either when the inadvertant actuation block (IAB) clears or when containment or reactor coolant system (RCS) level setpoints are reached. According to 10 CFR Part 50.46 (iii)(5),

"After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

In addition, SRP Section 15.4.8 indicates that GDC 28 provides assurance that the capability to bring the reactor to a safe shutdown condition will not be impaired by a control rod ejection accident.

As described in FSAR 15.4.8.3.2, conservative scram characteristics are applied as the~~,~~ "highest worth [control rod assembly (CRA)] (other than the ejected rod) remains stuck out of the core." While the effects of the ejected rod and fully stuck rod are addressed in the short term, it is unclear if the long term effects have been evaluated.

The staff notes that NuScale has requested an exemption to GDC 27, as documented in SECY 18-0099 (ADAMS ML18065A431). SECY 18-0099 states that the staff intends to apply demonstration that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded for postulated accidents that result in a return to power as the part of the technical basis for reviewing the GDC 27 exemption. However, the return to power analysis in FSAR 15.0.6 only evaluates a single stuck rod and not the combined loss of reactivity associated with an ejected rod and stuck rod.

To demonstrate that the minimum critical heat flux (CHF) in FSAR 15.0.6 analyses (which is the long-term analysis that applies to all Chapter 15 events that can result in a return to power) bounds the scenario where ECCS actuates due to the ejected rod and stuck rod, the staff is requesting justification that the current return to power analyses in FSAR 15.0.6 bound this scenario, at any time in the cycle. The justification should address all effects that could either reduce shutdown margin or lead to a return to power, including core boron dilution (see RAI 8930, effects from soluble boron plate-out, boron lost to the lower part of containment and diluted condensate return to the reactor pressure vessel via the reactor recirculation valves, etc.).