Enclosure 1:

"NuScale Accident Source Terms Methodology Options" PM-0818-61166, Revision 0

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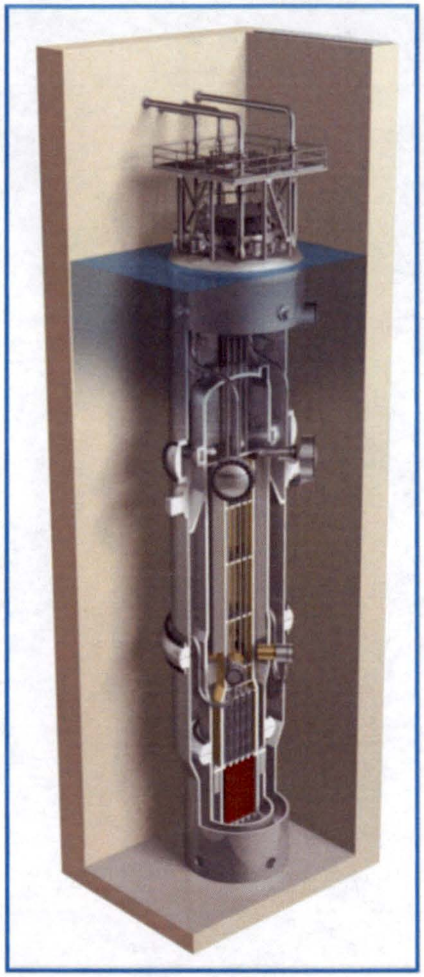
NuScale Accident Source Term Methodology Options



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Outline

- Purpose
- Background
- Overview of accident source term methodology options
  - Options to revise characterization or implementation of the maximum hypothetical accident (MHA) analysis within the NuScale licensing basis
  - Five total options, binned into two general groups
- Summary

Purpose

- Continue discussion of NuScale's proposed accident source terms methodology
- Present additional options to address MHA to resolve post-accident sampling and environmental qualification issues
- Receive NRC feedback on various options presented

Background

- Original NuScale Accident Source Term Methodology Topical Report (AST LTR) led to the use of a maximum hypothetical accident (MHA) that was overly conservative
  - Deterministically assumed significant core damage, which becomes design basis for certain aspects of plant design
  - Post-accident sampling (PAS) doses and environmental qualification (EQ) of in-containment instruments began to drive possible design changes without identifiable safety benefit
  - **Notes:**
    - "MHA" is shorthand for the scenario described in 10 CFR 52.47(a)(2)(iv), assumed for evaluating public doses at site boundary
    - NuScale relies on the normal process sampling system to serve as the post-accident sampling system (PASS)

Background

- NuScale proposed a revised methodology to develop MHA in "Accident Source Terms Regulatory Framework White Paper" (WP-0318-58980)
  - Credibility of core damage event informs selection of MHA
  - MHA and deterministic event sequences together comprise design basis radiological releases
- NuScale refined that proposal based on NRC feedback in previous public meetings and added a similar approach – Engineering Analysis (Option 1A below)
- NuScale developed other ways to resolve licensing challenges associated with core damage MHA

Regulatory Framework

- Offsite dose analysis, 10 CFR 52.47(a)(2)(iv)
  - Applicant is required to "assume a fission product release from the core into the containment," to determine that resulting offsite doses are within acceptable limits
    - Analysis considers leakage from an intact containment, effects of fission product mitigation systems, and postulated site parameters
    - This assumed fission product release is the "maximum hypothetical accident"
      » "…a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events…"
      » Original footnote further states "result[s] in potential hazards not exceeded by those from any accident considered credible"
  - Therefore, MHA is historically rooted in worst "credible" accident

Regulatory History

- MHA was originally defined by TID-14844 in 1962
  - Generalized hypothetical scenario for site evaluation purposes
    - Derived from accidents deemed credible for very early reactor designs
    - Posited event was "arbitrary," with "loose" tie to a major LOCA; nonmechanistic assumptions based on qualitative notion of credibility
  - ACRS acknowledged "for certain reactors … judgment will indicate that the generalized accident is too severe"
    - In 1963 AEC accepted a siting analysis reducing the assumed core melt from 100% to 6% based on crediting emergency core cooling system (ECCS) performance
    - General Design Criteria, and later 10 CFR 50.46, rendered core melt LOCA beyond the ECCS design basis
    - But "substantial core melt" persists as normal practice (RG 1.183)

Regulatory History

- MHA as design basis
  - Original role of MHA was to define an exclusion area and low population zone – i.e., separation from public as defense in depth
  - Larger reactors could not meet dose limits without crediting engineered safety features (ESFs) to mitigate doses
  - MHA became design basis for those ESFs and supporting SSCs (e.g., control room habitability, instrumentation), despite being beyond design basis for ECCS
  - 1996 Part 100 rulemaking moved requirement to Part 50

Regulatory History

- NuScale believes it is appropriate to reconsider core damage MHA assumption as design basis
  - 1996 Part 100 rulemaking: "events having the very low likelihood of about 10^{-6} per reactor year or lower have been regarded in past licensing actions to be 'incredible', and as such, have not been required to be incorporated into the design basis of the plant"
  - 1999 Alternative Source Terms rulemaking: "there is no regulatory requirement for a specific source term for reactors to be licensed in the future"

Overview of Options

- There are two categories of options and several possible technical/regulatory variations within each group
  - Group 1 options: Evaluation of design basis accidents, defense in depth, and risk insights to determine MHA source term
  - Group 2 options: Limit application of core damage MHA in the design basis
- Under all options:
  - Doses to public are shown acceptable for spectrum of postulated events
  - Emergency preparedness remains last layer of defense in depth
  - No significant differences in design
  - Compliance with 10 CFR 50.34(f)(2)(xix), core damage accident monitoring, will be addressed through the approach described in WP-0318-58980 (applying Equipment Survivability policy)

Option Group 1 Overview

- Evaluation of design basis accidents and defense in depth to determine MHA source term
  - Option 1A: Chapter 15 engineering analysis
  - Option 1B: White Paper methodology with expanded risk insights discussion

- Both options:
  - Address MHA consistently and holistically in NuScale design basis
  - Do not require exemptions to implement

Option 1A: Chapter 15 Engineering Analysis

- Use iodine spike as MHA source term if:
  - Chapter 15 design basis accidents do not result in fuel failure, AND
  - design incorporates defense in depth, AND
  - Chapter 19 confirms likelihood of severe accidents is very small

- Otherwise, use core damage MHA source term

Option 1A – Evaluate DBAs

- NuScale Chapter 15 design basis accidents do not result in fuel failure
  - The design eliminates many postulated accidents (large break LOCAs)
  - The design increases reliability (natural circulation versus active systems, no reliance on electric power or nonsafety support systems, no operator actions credited)
  - AOOs, infrequent events, postulated accidents, and special events use critical heat flux (CHF) as acceptance criteria:
    - CHF more restrictive than fuel failure (LWR)
    - Chapter 15 accidents do not result in core uncovery; maintaining primary inventory ensures core cooling

Option 1A – Evaluate DID

- NuScale design incorporates layers of defense in depth; defense in depth also accounts for uncertainties:
  - Prevention:
    - Simple, passive design eliminates numerous systems and components of principal concern to large light water reactors (LLWRs) – reduced operational challenges/transients
  - Containment:
    - Redundant, diverse, passive safety systems require no AC power or operator actions
    - Small core size (5% LWR) with low power density and high ratio of RCS to core power – increased core coolability.
    - Additional redundant design features support core cooling (CVCS, CFDS).
  - Protection:
    - Reactor pressure vessel (RPV) & containment vessel (CNV) design minimizes challenges to containment integrity (no penetrations below fuel, vessel-in-vessel design, partially submerged in UHS, no concrete).
    - Seismic Category I reactor building.
  - Overall robust plant design – minimize challenges and survive hazards.

Option 1A – Confirm Severe Accident Risk is Very Small

- NuScale Chapter 19 confirms risk from severe accidents is very small:
  - Very low NuScale PRA core damage frequency (CDF) provides significant margin to NRC's safety goal
  - Very low NuScale PRA large release frequency (LRF) provides significant margin to NRC's safety goal and minimizes dose to workers and the public

Option 1B: White Paper Framework

- Utilize both deterministic analysis and risk insights as proposed in "Accident Source Terms Regulatory Framework White Paper" (WP-0318-58980)
  - Option 1B has been expanded to address Staff concerns, which are addressed in subsequent slides:
    - risk-informed versus risk-based
    - PRA acceptability
    - PRA uncertainty
    - COL items and finality issues
    - external events

Option 1B – Risk-Informed

- A NuScale Power Module (NPM) is evaluated against the five key principles of RG 1.174 in a risk-informed decision making process to determine the appropriate MHA source term
  1. Current regulations are met, or are addressed through a requested exemption or rule change
  2. Safety margins are maintained
  3. Risk results are low and meet the Safety Goal Policy Statement and subsidiary objectives
  4. Performance measurement strategies are employed to confirm the selection of the MHA source term.
  5. Sufficient defense in depth exists (see next slide)
- Interdisciplinary team (Engineering, Licensing, PRA) makes final recommendation:
  - MHA iodine spike source term if principles and DID guidelines are met
  - Otherwise core damage source term

Option 1B – Defense in Depth

- As shown below, the proposed framework includes both quantitative and qualitative elements of likelihood and consequences to evaluate plant capabilities and programmatic elements

DID Layer	DID Guideline
Prevention	Maintain frequency of all design basis events $<10^{-2}$ per module critical year
Containment	Maintain frequency of all beyond design basis events $<10^{-4}$ per module critical year and individual sequences $<10^{-6}$ per module critical year[1]
Protection	Maintain large release frequency of all beyond design basis events $<10^{-6}$ per module critical year[2]
Overall	Ensure no single design or operational feature is exclusively relied upon to ensure any of the layers of defense

Notes:

1. The 10^{-4} per year threshold for core damage is the subsidiary objective derived to provide margin to NRC's safety goal. The 10^{-6} per year threshold is consistent with past licensing actions (61 FR 65176) and the State-of-the-Art Reactor Consequence Analysis for core damage sequences that did not lead to an early containment failure or containment bypass.
2. The 10^{-6} large release threshold is consistent with the performance guideline established by NRC to determine whether a level of safety ascribed by a reactor is consistent with the safety goals.it also supports compliance with regulations on dose to workers and the public.

Option 1B – Uncertainty & Scope

- Consistent with the guidelines in NUREG-1855, uncertainties incorporated into the NPM PRA and also addressed through the use of "mean" risk values; sources of uncertainty not resolved via quantification are addressed by other methods such as sensitivity studies, safety margins, and defense in depth
- The scope of events for MHA will include intact-containment core damage sequences from the PRA for all modes of operation (i.e., full power and low power and shutdown), including external events
  - The use of intact-containment beyond design basis events is consistent with 10 CFR 52.47(a)(2)(iv) in which accidents are hypothesized as releases into containment
- In addition the scope of events will include consideration of potential multi-module core damage sequences

Option 1B – PRA Acceptability & Monitoring

- The DC/COL-ISG-028 guidance addresses how DC and COL applicants can use the ASME/ANS Standard, as endorsed by RG 1.200, for determining PRA technical adequacy
  - Noted exceptions to meeting Capability Category II
    - generic versus plant-specific operating experience and data, and design and operational expectations versus plant-specific operating procedures and walkdowns
      - » The ISG includes enhancements and new requirements to capture the documentation of assumptions, uncertainties, and their impacts due to the status of the design, site, operational, and maintenance information and data
  - DC and COL PRAs should include self-assessments and external review
- PRA updates will include review of MHA source term selection:
  - When the PRA is performed for the COLA, site-specific information will be available, and the PRA will be updated to account for site-specific design information and any design changes
  - Peer review confirms acceptability of PRA for risk-informed applications

Option Group 2 Overview

- Assume core damage MHA, but limit application of MHA in the design basis
  - Option 2A: Analyze MHA only for offsite dose analysis only
  - Option 2B: Exclude MHA from design basis of PAM and PAS
  - Option 2C: Revise PAM and PAS capabilities under MHA
- All options:
  - Analyze a core damage MHA for offsite dose consequence evaluation
  - Address MHA challenges narrowly; does not fully account for "enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials," per 10 CFR 52.47(a)(2)(iii)
  - Require exemptions

Option 2A

- Analyze core damage MHA for offsite dose consequence purposes only
  - Approach:
    - MHA offsite consequences analyzed separately from other Chapter 15 events, does not constitute design basis event
    - MHA analysis satisfies 10 CFR 52.47(a)(2)(iv), remaining Chapter 15 events constitute design basis
    - EQ, CNV leakage, and control room dose depend only on "Category 1" source terms (no new "Iodine Spike MHA" event)
  - Considerations:
    - May require exemption from historical implementation of 10 CFR 52.47(a)(2)(iv) to define MHA as a BDBE
    - Does not resolve PAS challenges, additional exemption likely still required (see Option 2C)

Option 2B

- Exclude MHA from design basis of PAS and EQ
  - Approach:
    - Exemption from 10 CFR 50.49 to exclude MHA as an applicable DBA
    - Exemption from 10 CFR 50.34(f)(2)(viii) PASS rule that specifies MHA source term for sampling capability basis
    - Core damage MHA remains a Chapter 15 analysis associated with design basis of CNV leakage and control room dose
    - PAS capability demonstrated for non-core damage events only
  - Considerations:
    - Selective, inconsistent application of core damage MHA
    - Uncertain basis for non-core damage capability PAS and PAM

Option 2C

- Revise PAM and PAS capabilities under MHA
  - Approach
    - Revise EQ basis (e.g., FSAR Table 3.11-1) to reduce qualification duration for PAM
    - Instrumentation qualified to same dose levels as typical LWRs. PAM instrumentation qualified for duration actually required by operators for core damage event
    - Full exemption from 10 CFR 50.34(f)(2)(viii) PAS rule
  - Considerations
    - PAM approach conforms with regulations and guidance
    - Departure from precedent of longer PAM survival durations justified by NuScale's lack of required operator actions
    - PAS would be limited to contingency means for sampling, only

Conclusion

- NuScale continues to believe it is appropriate to revisit the assumption of core damage for MHA
- NuScale presented 3 additional options to more narrowly address design challenges associated with the core damage MHA

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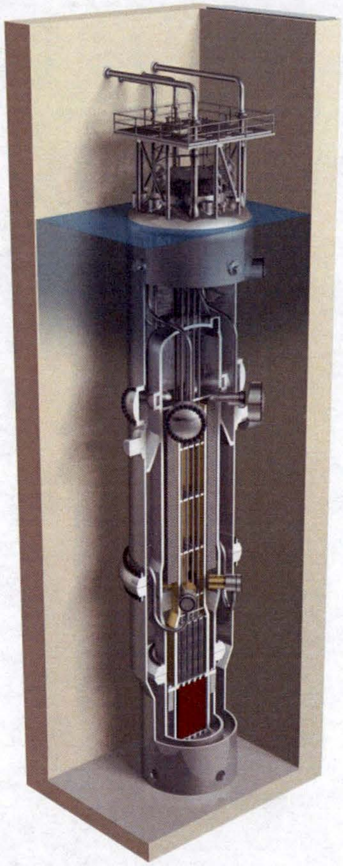

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