NuScale Accident Source Term Methodology

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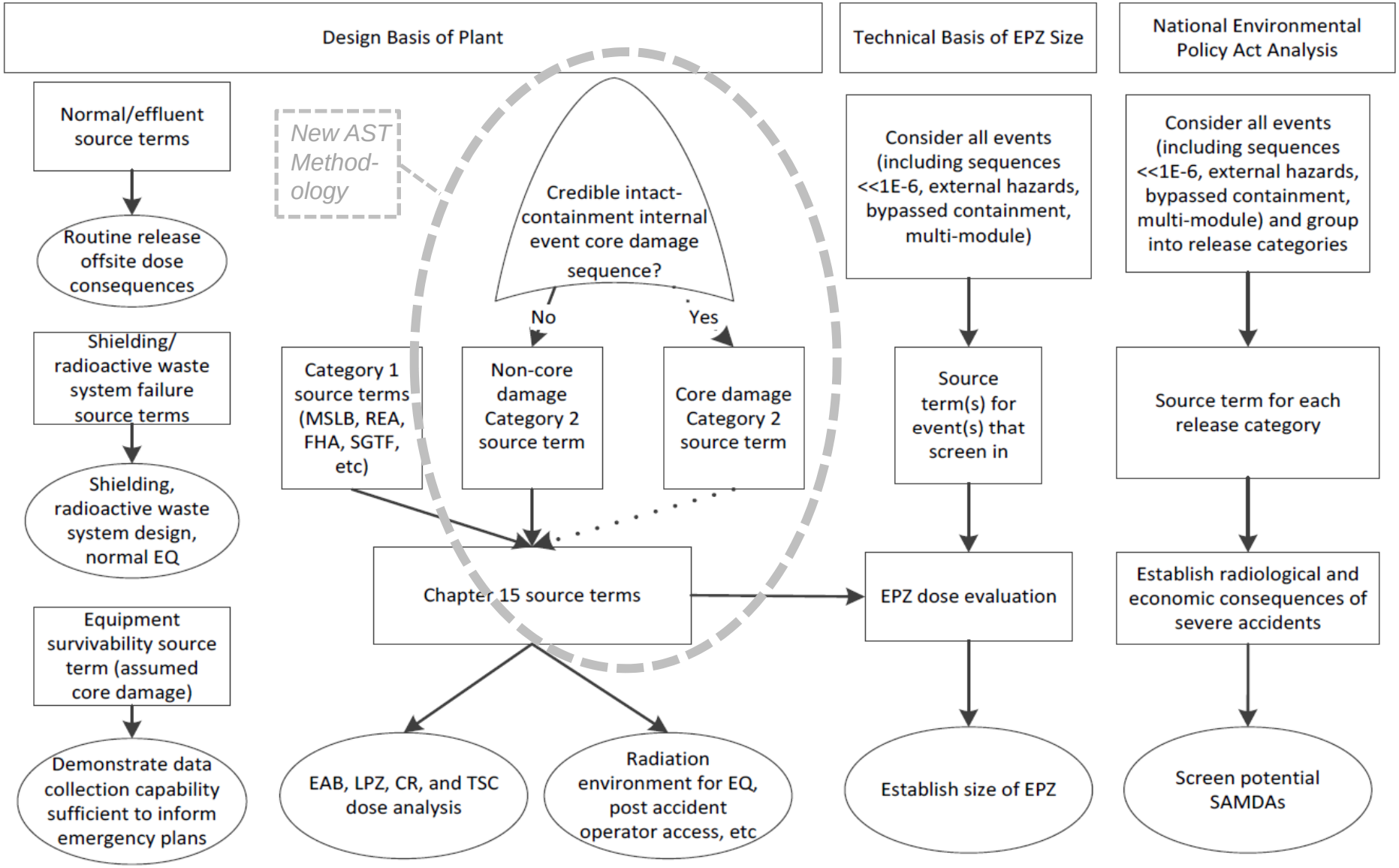
Purpose

- Discuss topics NRC expressed interest in at the NuScale Accident Source Term Methodology Topical Report (AST LTR) draft Revision 3 audit exit meeting related to the maximum hypothetical accident (MHA)

  Note: “MHA” is shorthand for the event evaluated pursuant to 10 CFR 52.47(a)(2)(iv)

- Discuss NRC feedback on Accident Source Terms Regulatory Framework White Paper (WP-0318-58980)

Overview of NuScale Source Terms



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Incredibility Threshold for MHA

- Intact-containment internal event core damage sequences with a frequency below 10^{-6} per year from the plant-specific PRA are classified as “incredible” for MHA.
- This 10^{-6} per year threshold has regulatory precedent.
  - 10 CFR Part 100 rulemaking: “It is worth noting that 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.”
  - The State-of-the-Art Reactor Consequence Analysis (NUREG-1935) used a screening criterion of core damage frequencies above 10-6 per year for scenarios with an intact containment, consistent with the role of the MHA in evaluating containment performance.
  - In Accident Sequence Precursor Program only events with conditional core damage probability or changes in core damage probabilities above 10^{-6} per year are categorized as precursors.

AST LTR Core Damage MHA

- If a credible, intact-containment, internal core damage event is identified, a core damage MHA is developed similarly to the current MHA approach discussed in AST LTR Rev. 2 that NRC already reviewed
  - The one exception is a change in aerosol modeling input to core damage MHA source term dose calculations in AST LTR Rev. 3
    - Aerosol modeling method updated to include diffusiophoresis with wall condensation, and thermophoresis
      - » Capabilities previously detailed in AST LTR, so few changes required (eRAI 8691 Question # 01.05-25 asked why these aerosol removal modeling capabilities were discussed but not credited)
      - » These models were originally excluded from methodology for simplicity
      - » Validation work for these models continued after original methodology submittal, and given the required rededication of STARNAUA, these models are now credited

AST LTR Iodine Spike MHA

- If no credible, intact-containment, internal core damage event is identified, an iodine spike MHA is developed. Iodine Spike MHA is a surrogate event based upon collective limiting T/H inputs and assumptions from all design basis events.
  - Assumes a generic failure resulting in release of all primary coolant from the RCS to CNV; this scenario is not a specific event or an extension of a specific event, therefore no event-specific T/H simulation data is available for use as input (or as leak rate basis) in the Iodine Spike MHA evaluation
  - However, evaluation of CNV pressure and temperature response to the most limiting break events using NRELAP5 concludes that CNV pressure reaches ½ peak pressure for the limiting AOO case in 35 minutes, at which point the system is in long term cooling and temperature and pressure steadily decline
  - Respective of this T/H profile, the RG 1.183 leak rate assumptions are considered conservative for the Iodine Spike MHA event

COL Implementation

- Site specific PRA model will confirm applicability of Iodine Spike MHA source term for COL application.
  - UFSARs will confirm applicability every 24 months
  - Upgraded site specific PRA will be produced every 4 years
- If an applicant or licensee identifies an event above the threshold, then the applicant or licensee would further evaluate credibility of event or incorporate design features to mitigate the consequences of the event.

Regulatory Assessment

- MHA originates from the offsite dose consequence evaluation requirement (aka siting analysis)
  - Core damage MHA has historically “flowed down” to other requirements because of its role as design basis for fission product release mitigation features
- Excluding core damage event from design basis affects NuScale implementation of numerous regulations vis-à-vis historical practice
- NuScale approach conforms with regulatory requirements
  - No exemptions appear necessary to implement
  - However, approach is a departure from some precedent and guidance

Regulatory Assessment

- Offsite dose analyses
  - Consistent with 10 CFR 52.47(a)(2)(iv)
    - Iodine Spike MHA source term represents an ”assume[d] fission product release from the core into the containment,” for which NuScale evaluates performance of fission product mitigation features to determine “offsite radiological consequences” within limits
    - Meets fundamental intent of MHA: bounds credible inside-containment accidents
    - “Major accident” and “core damage” assumptions described in footnote are historical practice, not strictly mandatory
  - Category 1 source terms show acceptable offsite doses for other deterministic events (including accidents outside containment)

Regulatory Assessment

- Control Room Habitability (GDC 19)
  - Evaluating category 1 and 2 source terms assures acceptable operator doses “under accident conditions, including loss-of-coolant accidents”
  - Complies with letter and intent of rule by ensuring habitability for all design basis accidents
- Environmental Qualification (10 CFR 50.49)
  - EQ based on Category 1 and 2 source terms addresses “radiation environment associated with the most severe design basis accident during or following which the equipment is required”

Regulatory Assessment

- 10 CFR 50.34(f)(2)(vii), (viii), (xxvi), and (xxviii)
  - All include "accident source term[11]" where Footnote 11 provides "the fission product release assumed for these calculations should be based upon a major accident…"
  - As with similar offsite dose rule, Iodine Spike MHA addresses "potential hazards not exceeded by those from any accident considered credible"
    - Literally complies with requirement, so exemption not necessary
  - But intent of requirements was expressly directed at reducing risk from severe accidents
    - TMI guidance reflects design and expected response of LLWRs
    - NuScale instead addresses risk in ways specific to NuScale design (e.g., prevention and containment); residual risk addressed by Emergency Planning

Regulatory Assessment

- 10 CFR 50.34(f)(2)(xix)
  - Rule explicitly addresses “monitoring plant conditions following an accident that includes core damage”
  - NuScale implementation addresses potential core damage events for this rule
    - Meets letter and intent of requirement
  - NuScale addresses through implementation of Staff’s “Equipment Survivability” position
    - Instrumentation to meet this rule is only necessary for severe accident response, as core damage event is not design basis
    - Differs from normal approach of meeting rule by implementing RG 1.97

Regulatory Assessment

- Equipment Survivability
  - Staff position intended to ensure equipment required for severe accident (SA) mitigation can survive in its environment
  - NuScale SA response is simplified: maintain containment integrity, monitor certain variables
  - Survivability of certain instrumentation for monitoring plant status during SA will be addressed in FSAR 19.2.3.3.8
    - Primary role for monitoring is support of emergency response, rather than mitigative actions
    - Meets 10 CFR 50.34(f)(2)(xix)

Regulatory Assessment

- Emergency Planning and Response
  - Approach reflects relationship between offsite dose limits and Emergency Planning (EP)
    - EP continues to serve as additional layer of defense-in-depth
  - No direct impact on proposed EPZ sizing methodology
    - For conservatism and consistent with intent of EP, methodology requires assessment of at least one core damage accident sequence
  - Equipment Survivability ensures functionality of instrumentation that supports the emergency response
  - Technical Support Center is demonstrated habitable for Category 1 and 2 accident source terms

AST LTR Content Feedback

- Does NRC request portions of the White Paper to be incorporated in AST LTR Revision 3?
  - NuScale does not plan to add such White Paper based scope to the AST LTR draft Revision 3, as seen by NRC in the audit.
- Does NRC agree with NuScale's identification of example results that would be updated in AST LTR Revision 3?
  - NuScale does not plan to update some examples because there is no value added to the review by updating them and it requires significant effort to update them.

Summary

- Discussions related to the White Paper and AST LTR Rev. 3 occurred
- RAI 9224 response, AST LTR Rev. 3, and associated DCA markups to be submitted July 2018

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