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U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of WP-0318-58980,"Accident Source Terms Regulatory Framework"

REFERENCES:

1. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of Presentation Materials Entitled 'NuScale Source Term Revision,' Revision 0, PM-0118-58201" (ML18019A163)
2. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of Topical Report TR-0915-17565, 'Accident Source Term Methodology,' Revision 2", dated September 11, 2017 (ML17254B068)
3. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of the NuScale Standard Plant Design Certification Application, Revision 1", dated March 15, 2018 (ML18086A090)

NuScale Power, LLC (NuScale) is proposing a new approach to the development of radiological accident source terms utilizing both deterministic analysis and risk insights. This new approach was discussed with the U.S. Nuclear Regulatory Commission (NRC) staff in a public meeting held January 22, 2018 (Reference 1). NuScale plans to incorporate the new approach in a forthcoming revision to the Accident Source Term Methodology Topical Report (Reference 2) and markups to the final safety analysis report (FSAR) associated with the NuScale design certification application (DCA) (Reference 3) by July 31, 2018.

The purpose of this submittal is to provide the regulatory framework for the new approach in a white paper entitled, "Accident Source Terms Regulatory Framework" (WP-0318-58980). Various radiological source terms are used in analyses associated with the design and licensing of the NuScale Power Module (NPM). The radiological source terms of principal interest to WP-0318-58980 are those used in design basis analysis associated with post-accident operator access, environmental qualification (EQ) of electric equipment, and the demonstration of the efficacy of fission product release mitigation systems under the scenario described in footnote 3 of 10 CFR 52.47 (a)(2)(iv) which NuScale refers to as the "maximum hypothetical accident (MHA)".

The most significant change in NuScale's new accident source term methodology is that overly conservative source terms based on "incredible" core damage events will be excluded from use as the MHA source term used in the design basis of the NPM. Specifically, intact-containment internal event core damage sequences with a frequency below 10^{-6} per year from the plant-specific probabilistic risk assessment (PRA) are classified as "incredible". The enclosed white paper describes why regarding $\leq 10^{-6}$ per year events as "incredible" is reasonable and consistent with past NRC licensing actions. The enclosure also elucidates the regulatory framework associated with a lack of a core damage event within the design basis of the NuScale plant. This regulatory framework would not require exemptions

to implement and utilizes both deterministic analysis and risk insights from the PRA to establish accident source term categories.

As the new source term methodology reduces the source term, resulting doses and other consequences are also reduced. Thus, staff conclusions on the current source term in the DCA remain valid, albeit with greater margin to regulatory limits. This should minimize additional review time and preclude schedule impact.

On the basis of the safety of NuScale's design and several analogous arguments to those presented in the enclosure, a case could be made that there are no "credible" accident sequences that would constitute the basis for emergency planning zone (EPZ) sizing. For conservatism and considering that the purpose of the EPZ is to plan protective actions to avoid or reduce dose to the public from a full spectrum of accidents, including beyond design-basis accidents, NuScale's EPZ sizing methodology that is currently under review requires assessment of at least one core damage accident sequence, in order to assure adequate public protection. The MHA source term is evaluated in NuScale's EPZ sizing methodology and there is no requirement on the magnitude or severity of the MHA source term. Any revision to the MHA source term is not expected to materially impact NuScale's EPZ sizing methodology or its review.

NuScale's new accident source term methodology may be of generic interest to other applicants. However, NuScale respectfully requests the proposal be evaluated specifically in the context of NuScale's design in order to facilitate review completion within the current DCA review schedule.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Jennie Wike at 541-360-0539 or at jwike@nuscalepower.com.

Sincerely,

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Enclosure: "Accident Source Terms Regulatory Framework White Paper", WP-0318-58980, Revision 0

Enclosure:

"Accident Source Terms Regulatory Framework White Paper", WP-0318-58980, Revision 0

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ACCIDENT SOURCE TERMS REGULATORY FRAMEWORK

1.0 Purpose

The purpose of this white paper is to describe NuScale's new accident source term methodology and evaluate its implementation within the context of the regulatory framework and historical precedents.

2.0 Background

NuScale initially introduced the new accident source term methodology to Nuclear Regulatory Commission (NRC) staff in a public meeting held January 22, 2018 (Reference 8.1). NuScale plans to formally incorporate this methodology into the NuScale design certification application (DCA) (Reference 8.50) by July 31, 2018.

This white paper is specific to the NuScale small modular reactor (SMR) design, as the paper relies on design-specific information. It is not intended to generically address an accident source terms framework that would be applicable to other designs.

The radiological source term of principal interest to this white paper is that used in demonstrating the efficacy of fission product release mitigation systems under the scenario described in Footnote 3 of 10 CFR 52.47(a)(2)(iv), which NuScale refers to as the "maximum hypothetical accident" (MHA).

The name "MHA" implies that it is the most bounding source term in all circumstances, however a better description would be that the MHA is the maximum hypothetical fission product release into an intact containment, postulated as a design basis event. The MHA source term has historically assumed a non-mechanistic substantial core melt event due to a major loss of coolant into an intact containment, which due to its severity was expected to bound the radiological consequences of all other accident source terms. The selection of the MHA impacts the source terms considered in designing and evaluating such features as control room habitability, post-accident operator access, and environmental qualification (EQ) of electric equipment.

In the new NuScale accident source terms methodology, an MHA is selected that also bounds all credible fission product releases inside containment, and allows an MHA source term without core damage provided there is no credible event that would lead to core damage.

Evaluation of severe accident mitigation design alternatives (SAMDAs), emergency planning zone (EPZ)[1], radioactive waste system, and effluent related source terms are not impacted by NuScale's new accident source term methodology. These source terms are discussed in this white paper only to provide context.

[1] The MHA source term is evaluated in NuScale's EPZ sizing methodology and there is no requirement on the magnitude or severity of the MHA source term. Any revision to the MHA source term is not expected to materially impact NuScale's EPZ sizing methodology or its review.

3.0 Regulatory Framework: Requirements and Guidance Related to Source Term

3.1 Accident Source Term Requirements Overview

The primary regulation of interest to the implementation of NuScale's new accident source term methodology is the offsite dose evaluation required by 10 CFR 52.47(a)(2)(iv), which utilizes the MHA source term. At a high level, the MHA source term originated as a siting requirement in 10 CFR 100.11, which defined the necessary exclusion area and low population zone for a reactor design and thus aided in evaluating site suitability. The siting evaluation was necessarily imprecise because it was performed at the Construction Permit application stage when only preliminary design information was available. Thus, Atomic Energy Commission (AEC) staff assessed the early reactor license applications (pre-1961) to develop a broadly encompassing "maximum credible accident" for future applicants to use for siting purposes (TID-14844, Reference 8.4).

Although reactor designs evolved to include safety features designed to prevent major core damage, the core melt MHA source term continued to be assumed. As the maximum postulated fission product release, the core melt MHA source term became the design basis for engineered safety feature (ESF) fission product mitigation systems and in turn for supporting and associated functions like control room habitability and EQ. When the Three Mile Island (TMI) Action Plan (References 8.36 and 8.55) directed actions to reduce the risk associated with severe accidents by bolstering monitoring, mitigation, and response capabilities, the core melt MHA source term served as an established severe accident radiological dose for which performance of those features could be demonstrated.

Table 3-1 presents a summary of pertinent regulatory requirements and associated history for accident source term determination, which are discussed in greater detail in Appendix A. Section 6 addresses how these requirements are satisfied if the MHA does not consist of a core damage event.

Table 3-1. Summary of Regulatory Requirements

Requirement	Summary	History/Intent
10 CFR 52.47(a)(2)(iv), offsite dose evaluation	Analyze an assumed fission product release into containment to determine offsite radiological consequences are within acceptable limits. Footnote: "The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. These accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products."	Originated as a siting requirement in 10 CFR 100.11, which defined the necessary exclusion area and low population zone for a facility based primarily on the core size and site conditions. As reactors became larger, fission product mitigation systems eventually became the dominant factor in meeting the dose limits. In 1997 the requirement was moved to final safety analysis report (FSAR) requirements (10 CFR 50.34) to reflect its primary role as ESF design basis. TID-14844 (1962) was developed to loosely encompass the postulated releases evaluated in early license applications. Reflected a realistic appraisal of the consequences of all "significant and credible fission release possibilities." Yielded a "pipe rupture-meltdown sequence … not likely to be exceeded by any other 'credible' accident."
Standard Review Plan (SRP) Chapter 15, offsite dose evaluations	Various design basis radiological events are evaluated against acceptance criteria derived from Part 100 (i.e., 10 CFR 52.47(a)(2)(iv)) offsite dose limits.	Provides reasonable assurance design basis radiological events result in acceptable offsite doses based on the severity of the event.

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Requirement	Summary	History/Intent
General Design Criterion (GDC) 19, control room habitability	"A control room shall be provided from which actions can be taken" under accident conditions. "Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."	Provide reasonable assurance of operator safety while performing actions to mitigate accidents, such that the operator actions are not inhibited. The core melt MHA source term was considered credible for early designs. The core melt MHA source term became the design basis for ESF fission product mitigation systems, and thus for the safety of operators necessary to provide reasonable assurance of the functionality of those systems.
10 CFR 50.49, environmental qualification	Design certification applicants must identify, and license applicants must establish a program for qualifying, "electric equipment important to safety," including safety-related and certain nonsafety-related electric equipment, and "certain post-accident monitoring equipment." The qualification program must include "the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional."	Rule development began before, but concluded after, the TMI accident. The "safety-related electric equipment" relied upon for design basis events includes systems (e.g. fission product mitigation ESFs and supporting features) for which the design basis was the MHA, so they were required to be qualified for the MHA. Includes qualification of "certain post-accident monitoring" (PAM) Regulatory Guide (RG) 1.97 (Reference 8.15) addresses that purpose as well as several of the TMI Action Items concerning severe accident monitoring.
10 CFR 50.34(f)(2)(vii), shielding for vital access and safety equipment 10 CFR 50.34(f)(2)(viii), post-accident sampling	"Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term[11] radioactive materials…" "Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term[11] radioactive materials" without exceeding specified worker dose limits.	*TMI Items II.B.2 and II.B.3*. Per NUREG-0660 (Reference 8.36), these Items were short term actions intended to "enhance public safety" by reducing risk from core degradation accidents, which can lead to containment failure and large releases. NUREG-0737 (Reference 8.55) specified an assumption of the RG 1.3 (Reference 8.53) or RG 1.4 (Reference 8.54) source term (based on TID-14844) as the accident conditions for demonstrating these severe accident capabilities.

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Requirement	Summary	History/Intent
10 CFR 50.34(f)(2)(xix), core damage monitoring	"Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage."	*TMI Item II.F.3*. The II.F Items aimed to "provide instrumentation to monitor plant variables and systems during and following an accident." Item II.F.3 concerned instrumentation to support "unplanned action if…a safety system is not functioning" and "action necessary to protect the public and for an estimate of the magnitude of the impending threat." The requirement was implemented via RG 1.97, which addresses the "expanded ranges" and "damaged core" source term to provide reasonable assurance of those capabilities.
10 CFR 50.34(f)(2)(xxvi), leakage control outside containment 10 CFR 50.34(f)(2)(xxviii), control room leakage	"Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term[11] radioactive materials following an accident." "Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term[11] release, and make necessary design provisions to preclude such problems."	*TMI Item III.D.1.1 and III.D.3.4.* The III.D Items addressed "design features that will reduce the potential for exposure to workers at nuclear power plants and to offsite populations following an accident." Item III.D.1.1 was a radiological release "source control" measure, that required licensees to reduce leakage to the extent practical "for all systems that could carry radioactive fluid outside of containment," without regard to a particular source term. Item III.D.3.4 was part of worker radiation protection improvements "to allow workers to take effective action to control the course and consequences of an accident." The subsequent rulemaking included a core damage source term in both requirements to address potential new leakage paths from the addition of, and provide reasonable assurance of control room habitability to operate, severe accident mitigation features.

Requirement	Summary	History/Intent
10 CFR Part 50 Appendix E.IV.E.8, Technical Support Center (TSC)	A licensee must provide and describe in their emergency plan an "onsite technical support center and an emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency."	TMI Item III.A.1.2 identified the need for upgraded emergency response facilities (ERFs) to improve the "inadequate" state of emergency planning and preparedness. Specified a TSC as "a place for management and technical personnel to support reactor control functions, to evaluate and diagnose plant conditions, and for a more orderly conduct of emergency operations." NUREG-0696 (Reference 8.18) describes the functional criteria for ERFs, including the TSC, including dose criteria equivalent to the control room.
Equipment Survivability	Severe accident "mitigation features must be designed so there is reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed."	The Severe Accident Policy Statement (Reference 8.37) provided that new power plant designs must address severe accident concerns by demonstration of compliance with the TMI requirements and consideration of severe accident vulnerabilities exposed by a PRA. SECY-90-016 (Reference 8.24) addressed the qualification of severe accident mitigation features to perform their functions. Staff concluded that features provided for only severe accident protection were not subject to safety-related requirements such as 10 CFR 50.49 environmental qualification and Appendix B quality assurance.
Note: The footnote in Table 3-1 is referring to footnote 11 in 10 CFR 50.34.		

4.0 NuScale Design Overview

The NuScale design achieves large safety margins and low risk through the use of small nuclear fuel inventories and simplified, passive safety systems that automatically actuate on a loss of power. The small nuclear fuel inventory (approximately 5% of the nuclear fuel of a traditional 1,000 MWe light water reactor (LWR)) and the lower power density significantly reduce the magnitude of the heat that needs to be removed to cool the core. Additionally, the initial ratio of RCS inventory per MWth of core power is approximately 4

times larger in the NuScale Power Module (NPM) compared to a traditional PWR, which is a key attribute that significantly increases the probability of successful core cooling under a wide range of challenges, and delays any event progression.

The safety related passive design features allow the NPM to safely shut down and self-cool with significant safety margins for an extended time without reliance on operator actions, AC or DC power, or additional water. Additionally, nonsafety-related active design features that allow low pressure coolant injection into the containment via the containment flooding and drain system and high pressure coolant injection into the RPV via the chemical and volume control system are two defense-in-depth capabilities for achieving core cooling.

NuScale's simple design eliminates numerous systems and components whose failures can contribute to core damage in traditional LWRs. Piping external to the reactor pressure vessel (RPV) is of short length and small diameter. There are no RPV or containment vessel (CNV) penetrations below the top of the reactor core. The integrated design of the NPM, encompassing the reactor, steam generators and pressurizer, and its use of natural circulation eliminates the need for large primary piping and reactor coolant pumps. Large-break loss-of-coolant accident (LOCA) scenarios, of principal concern in traditional LWRs, are precluded by the design of the NPM. The evacuated steel CNV allowed elimination of RPV insulation in the design, which prevents potential sump blockage concerns. Small-break LOCAs, extended station blackouts, and various other events of concern to traditional LWRs do not challenge the safety of the NuScale plant. The result is a design with a core damage frequency orders of magnitude lower than traditional LWRs.

The containment's primary function in the NuScale design is to retain reactor coolant for core cooling in a loss of coolant event, thereby preventing significant radiological release from occurring. The containment is also an essentially leak-tight barrier to release of radioactivity should an event occur. The reactor building housing the NPMs is a Seismic Category I structure designed to endure a broad seismic and severe weather envelope and protect the NPMs and spent fuel pool from aircraft impact. The reactor building is equipped with an HVAC filtration system. Within the reactor building, the high pressure steel CNV is mostly submerged in millions of gallons of borated water in the below grade reactor pool; providing passive cooling of the NPMs and spent fuel pool in excess of 30 days without a makeup water source or pool cooling and suppression of pressure inside the CNV. In the unlikely event of a severe accident, the vessel-in-vessel design would provide in-vessel retention of core debris. Additionally, the evacuated steel CNV excludes concrete cracking concerns seen in traditional LWR containments. Thus, unlike a traditional LWR, extended containment integrity does not depend upon successful emergency core cooling system (ECCS) performance.

The design addresses the safety and risk associated with multi-module plant operation. Except for the ultimate heat sink (UHS) reactor pool, safety-related systems are module-specific and functionally independent of shared systems and other NPMs. This independence precludes adverse interactions among NPMs as a result of safety-related system operation during design basis events. There is no design basis event that would pose a hazard to multiple NPMs. The operation of the nonsafety-related shared systems,

including postulated failures of these systems during design basis events, does not adversely affect safety-related NPM functions.

5.0 Accident Source Term Methodology

In 2012, the Nuclear Energy Institute issued a position paper on SMR source terms (Reference 8.5). NuScale participated in the development of this position paper. In this document, source terms were divided into two categories. “Category 1” consists of the deterministic suite of accidents. “Category 2” is the MHA.

The forthcoming NuScale topical report (TR-0915-17565 Revision 3, Reference 8.6) describes a generalized methodology for developing accident source terms and performing the corresponding radiological consequence analyses. The methodology conservatively develops the maximum radiological effects for Category 1 and 2 source terms. The methodology utilizes both deterministic analysis and risk insights to identify Category 1 and 2 source terms used in the design basis of the plant. A description of this approach is discussed in Section 6.

5.1 Category 1 Source Terms

The Category 1 source terms include standard deterministic accidents that are similar to those of large LWRs such as: main steam line break (MSLB), rod ejection accident (REA), fuel handling accident (FHA), steam generator tube failure (SGTF) and small primary coolant line break outside containment. NuScale’s Category 1 source term methodology is consistent with the RG 1.183 (Reference 8.7) methodology, which prescribes deterministic analyses.

5.2 Category 2 Source Term

The Category 2 source term is called the MHA. In order to conservatively bound all design basis event fission product releases, the MHA has generally been assumed to result in substantial meltdown of the core, and is used to evaluate the radiological consequences from a fission product release inside containment.

The NuScale methodology allows for the MHA source term to be developed in one of two ways. If the NuScale design does not result in any “credible” core damage events, a surrogate event that results in primary coolant entering the containment is used as the MHA source term. NuScale's methodology utilizes the term "Iodine Spike MHA source term" to refer to this scenario. If the NuScale design does result in a “credible” core damage event as defined in Section 5.2.1, a surrogate event that results in core damage is used as the MHA source term. NuScale's methodology utilizes the term "Core Damage MHA source term" to refer to this scenario. Since the NuScale design does not result in any credible core damage scenarios, the Iodine Spike MHA source term will be used for the NuScale DCA.

5.2.1 Risk Consideration for Core Damage Events

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While probabilistic risk assessments (PRAs) generally include frequent, infrequent, and rare events, accident frequencies can range to numbers that are extremely small (e.g., 10^{-9}/year). As such, distinguishing "credible" core damage events from "incredible" core damage events as evaluated in the PRA is a key aspect of NuScale's new accident source term methodology.

In the development of the MHA source term, the NuScale approach employs a 10^{-6} per year threshold for identifying incredible core damage events. Specifically, intact-containment internal event core damage sequences with a frequency below 10^{-6} per year from the plant-specific PRA are classified as "incredible." Sequences from all modes of operation (i.e., full power and low power and shutdown) are considered.

The frequency of 10^{-6} per year is comparable to the WASH-1400 estimation of the expected frequency of a large LOCA with core damage.[2] As discussed in Appendix A, TID-14844 reflected Staff's assessment of a maximum credible accident for early designs that lacked proven emergency core cooling capability. Thus, the WASH-1400 estimate is a reasonable indication of the likelihood (i.e. credibility) of an event sequence for which the MHA radiological release encompassed.

The selection of 10^{-6} per year as the threshold is consistent with the NRC Accident Sequence Precursor Program in which only initiating events with a conditional core damage probability or only changes in core damage probabilities above 10^{-6} per year are categorized as precursors to potential core damage accidents. It is also consistent with the State-of-the-Art Reactor Consequence Analysis in which only sequences with core damage frequencies above 10^{-6} per year were considered in the analysis (Reference 8.8).

Consistent with NRC safety goals and subsidiary objectives for reactors (i.e., a CDF less than 10^{-4} per reactor year and a LRF less than 10^{-6} per reactor year), core damage sequences with frequencies below 10^{-6} per year in the NuScale design are judged to have sufficient safety margins to account for uncertainties. In addition, the selection of 10^{-6} per year as the threshold is consistent with past licensing actions; as stated in the federal register notice associated with the 10 CFR Part 100 rulemaking (Reference 8.9) "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 use of intact-containment sequences is consistent with 10 CFR 52.47(a)(2)(iv) in which the MHA is hypothesized as a release into containment. The use of internal events is consistent with the RG 1.183 large LOCA and consideration of design basis accidents as surrogates for evaluating the response of ESFs. In addition to events that involve an individual module, the scope of the PRA includes a bounding quantitative consideration of multi-module core damage events (i.e., full-power internal-initiating events that lead to core damage in more than one module).

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[2] The large LOCA core damage sequences in WASH-1400 result in containment failure, while the TID-14844 methodology non-mechanistically assumes an intact containment.

Consistent with RG 1.200 (Reference 8.10), the PRA used to identify credible core damage events meets pertinent acceptability requirements to provide reasonable assurance it is sufficient for use in regulatory applications. The NuScale PRA meets this objective as it was developed using a procedure based on the quality assurance requirements specified in RG 1.174 (Reference 8.11). The NuScale PRA meets or exceeds the quality expectations of the NRC for a nuclear power plant design during the design certification phase of NRC review. These expectations include a self-assessment and also include engaging the services of an independent PRA expert panel to provide a high-level, expert-level review of the PRA.

6.0 Implementation

6.1 Summary of Source Term Implementation

The Category 1 and Category 2 source terms together comprise the array of accident source terms considered for application to the various regulatory/design requirements summarized in Section 3 and detailed in Appendix A. With the exception of equipment survivability, each requirement is assessed against the most challenging source term for that purpose.[3] Equipment survivability is implemented in a manner that addresses severe accidents.

For example, the Category 1 source term from the fuel handling accident may produce the highest radiation environments in the reactor building area and thus be used for EQ of reactor building equipment, control room habitability considerations, and as the bounding offsite radiological consequence event. Meanwhile, the Category 2 source term may produce the highest radiation environments in the containment and therefore be used for EQ of instrumentation in the containment. Similarly, a leak in a line carrying primary coolant outside of containment may be the most limiting source term for consideration in post-accident operator access in a particular area.

A high level overview of NuScale implementation of source terms is given in Figure 6-1 to provide context.

[3] By definition, the 10 CFR 52.47(a)(2)(iv) analysis only considers the MHA. However, the other Chapter 15 offsite radiological consequence analyses address the Category 1 source terms for the same purpose.

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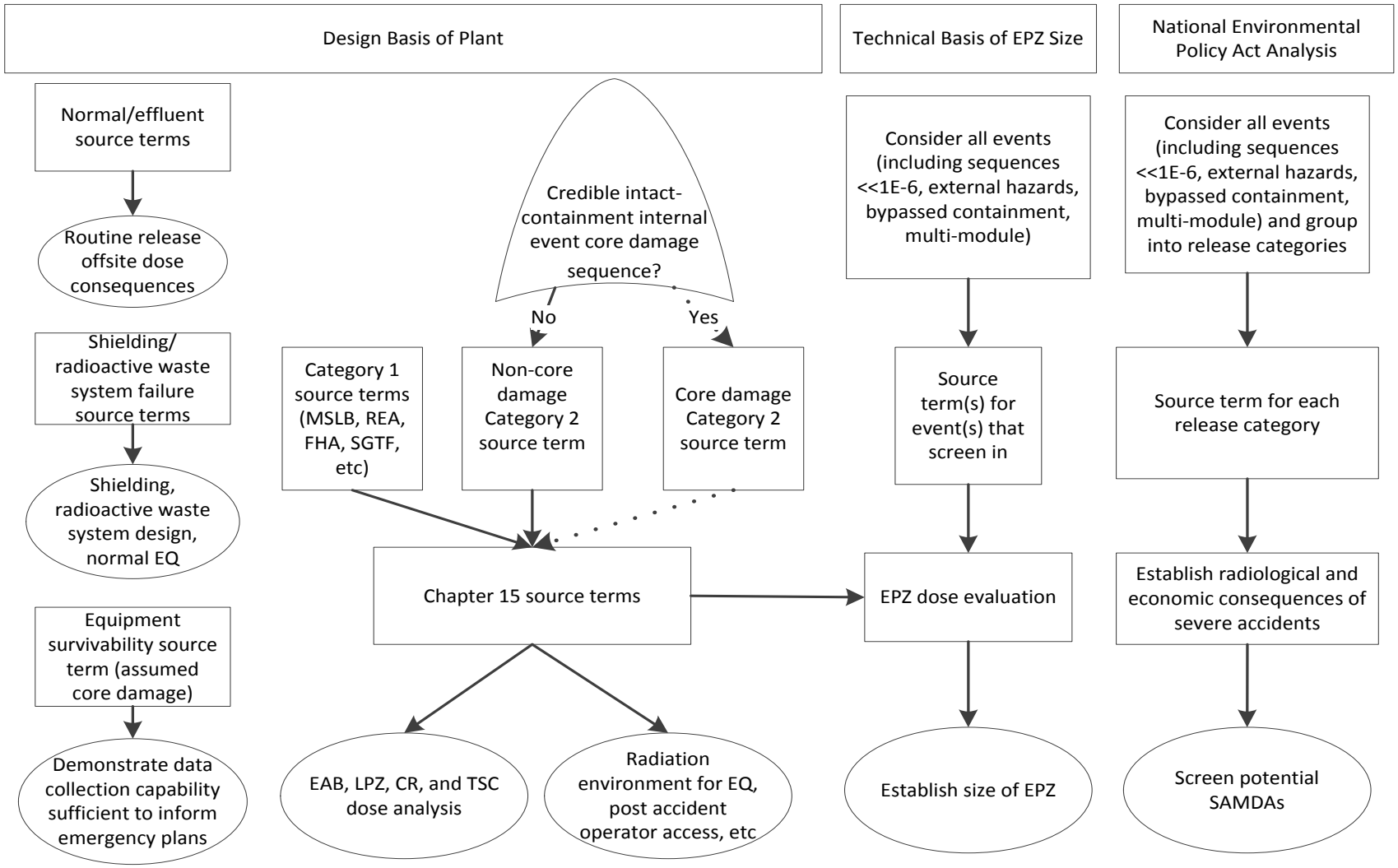

Figure 6-1. Overview of NuScale Source Terms

6.2 Regulatory Assessment

The following assessment of regulatory compliance assumes that NuScale's new accident source term methodology yields a non-core damage Iodine Spike MHA source term for the Category 2 source term. NuScale's implementation is assessed against the literal language and intent of the various requirements impacted.

6.2.1 Offsite Dose Requirements

As required by 10 CFR 52.47(a)(2)(iv), the Iodine Spike MHA source term represents an "assume[d] fission product release from the core into the containment," for which NuScale evaluates "the postulated fission product release…using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable postulated site parameters…to evaluate the offsite radiological consequences."

The Iodine Spike MHA source term fulfills the intent of evaluating the performance of fission product mitigation features to maintain offsite doses acceptably low in the event of a fission product release inside containment. The source term, as specified by the footnote to 10 CFR 52.47(a)(2)(iv), is "assumed for this evaluation" and "based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events." However, the Iodine Spike MHA source term departs from precedent in that it is not "assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products." Rather than assuming core melt, NuScale's approach is consistent with the historical recognition that for "certain reactors and conditions judgment will indicate that the generalized [MHA] is too severe" (Reference 8.31).

NuScale's approach is an extension of the historical trend towards reliance on ESFs, rather than physical separation (i.e. large distances to major population centers), to protect public health and safety from reactor accidents. Because of the reliability and performance of the NuScale ECCS, the "uncertainties associated with accident sequences and equipment performance at the time of promulgation" of 10 CFR 100.11 do not justify an overly conservative core melt assumption for the NuScale MHA (Reference 8.32). Under NuScale's approach, the "enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials," as provided by 10 CFR 52.47(a)(2)(iii), are considered.

While 10 CFR 52.47(a)(2)(iv) is directed at accidents inside containment, the Category 1 source terms complement the MHA evaluation by demonstrating acceptable offsite doses for deterministic design basis accidents outside containment, regardless of probability of occurrence, consistent with current SRP Chapter 15 practice. In this manner, the larger purpose of 10 CFR 52.47(a)(2)(iv), establishing a suitable site envelope to provide adequate offsite radiological protection, is fulfilled.

6.2.2 Control Room Dose and EQ Basis

The intent of these requirements is to provide reasonable assurance of control room habitability and electric equipment performance, respectively, for design basis events. Accordingly, the array of Category 1 and 2 source terms, including the Iodine Spike MHA, will be evaluated for conformance with GDC 19 and 10 CFR 50.49. This is consistent with the lack of credited operator actions or radiological release mitigation systems necessary to prevent or mitigate design basis events.

Because NuScale's implementation of the post-accident monitoring portion of 10 CFR 50.49 will address monitoring for Category 1 and 2 source term events only,[4] monitoring of severe accidents is addressed with respect to the relevant TMI requirements and equipment survivability as discussed below.

6.2.3 TMI Requirements

The TMI requirements encompass two groups of rules with respect to the language and intent of the requirements. The first group is 10 CFR 50.34(f)(2)(vii), (viii), (xxvi), and (xxviii), which require that an applicant address "source term radioactive materials," annotated by the source term footnote originally in 10 CFR 100.11. Applying the array of Category 1 and 2 source terms to these requirements meets the literal language of the requirements to consider source term radioactive materials under accident conditions, and is also consistent with the footnote to the extent discussed in Section 6.2.1. Unlike 10 CFR 52.47(a)(2)(iv), the TMI requirements footnote includes the provision, retained from original 10 CFR 100.11, that the postulated source term should result in "potential hazards not exceeded by those from any accident considered credible." Consistent with that statement, the Iodine Spike MHA source term represents an in-containment event that bounds the hazards of credible accidents not otherwise addressed by the deterministically postulated Category 1 accident source terms.

Unlike the offsite dose, EQ, and control room dose requirements, these TMI requirements were specifically directed towards severe accident response capability. Therefore, while NuScale's approach is consistent with the literal language of 10 CFR 50.34(f)(2)(vii), (viii), (xxvi), and (xxviii), fulfilling the intent of these rules requires addressing the underlying severe accident considerations. Per NUREG-0660, the TMI action items addressed "the reality of the risk, previously only theoretically assessed, of accidents that result in substantial degradation and melting of the core. This risk arises from the fact that core-degradation accidents can lead to containment failure and the eventual release of large amounts of radioactivity to the environment."

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[4] As discussed in Appendix A, compliance with RG 1.97 (Reference 8.15) is typically used to meet 10 CFR 50.49 as well as TMI requirements for additional accident monitoring capability.

NuScale's design yields a very low likelihood of severe accidents; the radiological consequences of those accidents are mitigated by passive ESFs rather than operator actions. Core degradation accidents do not lead to containment failure. Were a severe accident to occur in the NuScale design, the response would not depend on local operator actions informed by sampling and monitoring, as contemplated by 10 CFR 50.34(f)(2)(vii), (viii), (xxvi), and (xxviii). Because of these differences between the NuScale design and those designs considered when developing the TMI action items, it is appropriate to satisfy 10 CFR 50.34(f)(2)(vii), (viii), (xxvi), and (xxviii) by considering the Category 1 and 2 source terms, only.. Although a departure from precedent, this approach provides reasonable assurance of access and sampling capabilities for a range of accident conditions.

Unlike the four TMI requirements just discussed, 10 CFR 50.34(f)(2)(xix) explicitly requires "monitoring plant conditions following an accident that includes core damage." Therefore, to meet the literal language and intent of this rule, NuScale will provide monitoring capability that addresses severe accident considerations, without regard to the Category 2 source term selection. This implementation is addressed under the paradigm of equipment survivability, as discussed in Section 6.2.4.

6.2.4 Equipment Survivability

The NRC staff position on equipment survivability in SECY-90-016 is intended to reasonably assure that equipment required for severe accident mitigation can survive in the severe accident environment. For the NuScale design, severe accident mitigation is greatly simplified: the two functions of interest are maintaining containment integrity and certain PAM variables. Containment integrity reasonably assures that core damage events do not cause excessive doses to workers and the public. Since the NuScale design does not rely on credited mitigating operator actions, PAM is used to monitor plant status and support any required emergency response functions.

The containment performs both safety-related and severe accident functions. FSAR Chapter 6 addresses containment integrity for design basis conditions. In addition, Chapter 19 also addresses containment integrity for beyond design basis conditions. Accordingly, containment severe accident survivability is demonstrated.

By excluding a core damage MHA from the Chapter 15 event spectrum, implementation of RGs 1.89 (Reference 8.16) and 1.97 will not address the survivability of severe accident monitoring equipment (i.e., the severe accident source term assumed under RG 1.89 no longer applies). Accordingly, NuScale will apply the NRC staff position on equipment survivability in SECY-90-016 to reasonably assure certain instrumentation necessary to monitor critical safety functions following a severe accident would function for appropriate durations during and following a core damage accident. This instrumentation will not be the full array of design basis PAM variables. It is a specific set of instrumentation

sufficient to monitor plant status to support emergency response functions. This instrumentation will be identified in Section 19.2.3.3.8 of the FSAR. This approach also fulfills the language and intent of the severe accident monitoring capability required by 10 CFR 50.34(f)(2)(xix).

6.2.5 Emergency Planning

Development of emergency plans, including determination of the size of the EPZ, is outside the scope of the NuScale DCA. However, NuScale Topical Report TR-0915-17772 (Reference 8.17) provides a methodology for license applicants to establish a technical basis for the plume exposure pathway EPZ size for a NuScale plant. The EPZ methodology is not directly impacted by the new accident source terms methodology,[5] but NuScale recognizes that emergency planning and the offsite dose limits are related in their purpose of limiting doses to the public in the event of an accident. The EPZ methodology, with a wide frequency screening including a threshold based on a multiplier on the total CDF, is intended to produce an acceptable spectrum of accident sequences that would have consequences evaluated in order to determine EPZ size. Severe accident sequences are also evaluated for their potential to propagate to multi-module events, for possible inclusion in the EPZ technical basis.[6] In this manner, the NuScale EPZ will account for core damage sequences of substantially lower likelihood than the MHA, and thereby reasonably assure adequate public protection for those sequences.

With respect to structures, systems, and components (SSCs) that support the facility emergency response, the equipment survivability strategy discussed in Section 6.2.4 will provide reasonable assurance of a means for monitoring accident progression and the potential release of radioactive materials, as required by a licensee's emergency plan. The TSC will be demonstrated habitable for the same source terms as the control room, consistent with NUREG-0696.

6.3 Licensing Considerations

6.3.1 DCA Impacts

The forthcoming Revision 3 of the Accident Source Term Methodology Topical Report (TR-0915-17565) will present the methodology described in Section 5 of this white paper. NuScale FSAR Sections 3.11, 3.C, 12.2, and 15.0.3 of the DCA will utilize the applicable Category 1 or 2 source terms and will point to the methodology described in TR-0915-17565 Revision 3 instead of Revision 2 of TR-0915-17565 (Reference 8.19). Minor additions to wording in these FSAR sections will describe the bounding Category 1 or 2 source terms. A majority of the FSAR changes will be updates to numerical results in tables. NuScale FSAR Section 19.2.3.3.8 will present and justify the approach

[5] The MHA is one dose evaluated for the EPZ technical basis, but is not expected to be determinative.
[6] See NuScale Topical Report TR-0915-17772 for additional information.

summarized in Section 6.2.4 of this white paper. As the new source term methodology reduces the source term, resulting doses and other consequences are also reduced. Thus, staff conclusions on the current source term in the DCA remain valid, albeit with greater margin to regulatory limits. This should minimize additional review time and preclude schedule impact.

6.3.2 Combined License Implementation

The implementation of the accident source terms methodology will be reaffirmed by the combined license (COL) applicant and at specified intervals throughout the life of the plant on the basis of new information (e.g., site specific hazards, updated SSC failure rates, etc.). This will be accomplished through the requirement of 10 CFR 50.71(h)(1) to create a site specific PRA model, the requirement of 10 CFR 50.71(h)(2) to upgrade the site specific PRA every four years, and the requirement of 10 CFR 50.71(e) and 10 CFR 52.3(b)(6) to submit updated final safety analysis reports (UFSARs) at least once every 24 months throughout the life of the plant. For the latter, the COL FSAR or UFSARs would compare the upgraded site specific PRA results to the risk considerations presented in Section 5.2.1 to confirm that there is no credible core damage event for the Category 2 source term.

If an applicant or licensee identifies an event above the threshold, then the applicant or licensee would show through analysis that the event is incredible or incorporate design features to mitigate the consequences of the event.

7.0 Summary and Conclusion

As presented in this white paper, the regulatory framework supports NuScale's new accident source term methodology. NuScale utilizes both deterministic analysis and risk insights to identify the spectrum of source terms considered in establishing the design basis of the plant. Sufficient technical and regulatory basis exists to support the definition of incredible core damage events in the NuScale design for the purposes of establishing the MHA. This framework does not require exemptions to implement.

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8.5 Nuclear Energy Institute, "Small Modular Reactor Source Terms," December 27, 2012.

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8.16 U.S. Nuclear Regulatory Commission, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” Regulatory Guide 1.89, Rev. 1, June 1984.

8.17 Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites, TR-0915-17772.

8.18 U.S. Nuclear Regulatory Commission, “Functional Criteria for Emergency Response Facilities,” NUREG-0696, February 1981.

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8.22 U.S. Code of Federal Regulations, “Changes, Tests, and Experiments,” Section 50.59, Part 50, Chapter 1, Title 10, “Energy,” (10 CFR 50.59).

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Appendix A. Regulatory Requirements

A.1 Off-Site Dose Requirements

A.1.1 10 CFR 52.47(a)(2)(iv) Off-Site Dose Evaluation

The primary regulation at issue is the radiological consequences analysis required by 10 CFR 52.47(a)(2)(iv).[7] For this analysis, an applicant is required to "assume a fission product release from the core into the containment," and determine that the doses to an individual at the boundary of the exclusion area and low population zone are within acceptable limits. The analysis considers containment leakage from an intact containment, fission product mitigation systems, and postulated site parameters.

10 CFR 52.47(a)(2)(iv) originates from the siting analysis originally required by 10 CFR 100.11. That rule, as discussed below, provided guidelines for applicants in evaluating the acceptability of proposed sites in an effort to standardize the review performed by the AEC for early reactor sites. As captured in the footnote to 10 CFR 100.11, the evaluation typically assumed a major accident with core melt as a bounding assumption for the purpose of site evaluation:

> *The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.*

The Part 100 regulations were developed as general criteria to aid license applicants in evaluating their proposed sites during a time when reactor technology and plant sizes were rapidly evolving. The development of the Part 100 regulations stemmed from the AEC's conclusion that definitive criteria were needed to provide guidance to the public on the site review process.[8] The proposed 10 CFR 100.11 emphasized "environmental isolation" (i.e. separation) to afford public protection for accidents considered highly unlikely, with little consideration given to ESFs (Reference 8.27). However, the rule development coincided with licensing proposals for ever larger reactors nearer to population centers than in the past, which could not meet the siting criteria without "credit" for mitigating the radiological release through the use of engineered safety

[7] Equivalent requirements for COL applications are at 10 CFR 52.79(a)(1)(vi).
[8] "While recognizing the difficulty of writing detailed criteria at an early stage of a technology, it is in the interests of sound regulatory practice to have criteria to the extent possible to work from. In particular, the problem of the selection of suitable sites for nuclear reactors is and has been a troublesome point" (Reference 8.52).

features. Thus, the final rule allowed greater flexibility for "substitution [of] engineered safeguard[s]"[9] to protect public health and safety.

The source term assumed for the siting evaluation has a complex history reflecting licensing experience to that point. The plants proposed or built prior to 1961 either lacked or included unproven emergency core cooling systems, and thus a major core melt was postulated to result from a sudden large pipe break.[10] Moreover, the siting evaluation was performed as part of the Construction Permit process, for which only preliminary reactor design information was available. Thus, the guidance for applicants to perform the 10 CFR 100.11 siting analysis, published in Technical Information Document 14844 along with the final rule, reflected siting evaluations performed by the Advisory Committee on Reactor Safeguards (ACRS) to that point. Staff surveyed those simplified siting evaluations, which were based on the worst accidents considered credible for the specific designs, and developed a generalized hypothetical scenario for the purposes of future site evaluations. The accident assumed was an "arbitrary" one, "tied to a rupture of a major pipe," resulting in core melt. As the ACRS stated, "the reasoning back of this source term is admittedly loose," and "for certain reactors and conditions judgment will indicate that the generalized accident is too severe." (Reference 8.31)

As reactors grew in size and ESF designs matured, greater emphasis was placed on mitigating radiological release to compensate for large doses that would result from such a simplified siting evaluation. In addition to radiological consequence mitigation afforded by systems such as containment sprays and filtration, systems to reduce the source term release were sometimes credited. For example, in 1963 AEC staff and ACRS accepted a proposal to credit an ECCS as reducing the assumed core melt from 100% to 6% for the siting analysis.[11]

The shift in primary emphasis on facility ESFs rather than siting to mitigate public consequences from major reactor accidents was reflected when the siting dose evaluation became part of the Safety Analysis Report requirements in 10 CFR Part 50 in 1996,[12] and then subsequently in the 10 CFR Part 52 rules. The 10 CFR 100.11

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[9] "These guides and the technical information document are intended to reflect past practice and current policy of the Commission of keeping stationary power and test reactors away from densely populated centers. It should be equally understood, however, that applicants are free and indeed encouraged to demonstrate to the Commission the applicability and significance of considerations other than those set forth in the guides." (Reference 8.28)

[10] See, e.g., SECY-89-153: "From the outset, these in-containment source terms were widely acknowledged to be very conservative but were justified on the basis of the uncertainties associated with accident sequences and equipment performance at the time of promulgation (circa 1962)." (Reference 8.32)

[11] "Regulatory Staff noted this site could not tolerate 100% meltdown of the fuel and full release of the fission products to the containment. Credit had to be given for an emergency core cooling system, so that only 6% of the core was assumed to melt with reduced release of fission products to the containment....The ACRS accepted the approach." (Reference 8.33, p. 2-68)

[12] "The proposed criteria basically decouple siting from accident source term and dose calculations. Experience has shown that these factors have tended to influence plant design aspects rather than siting. Accident source term and dose considerations are proposed to be applied to plant design aspects and would be relocated to part 50." (Reference 8.34). Similarly, the GDCs as originally proposed in 1965 included specific reference to a major accident without core quenching as the design basis heat and pressure for containment integrity. That requirement was

requirements and its footnote were copied almost verbatim, with the addition of an explicit reference to crediting "fission product cleanup systems."

During the Alternative Source Terms (10 CFR 50.67) rulemaking, finalized in 1999, the phrase "would result in potential hazards not exceeded by those from any accident considered credible," was deleted from the 10 CFR 50.34 and Part 52 footnotes. The modified footnote provided reasonable assurance that new design applicants could implement the revised source term methodology of NUREG-1465 (Reference 8.30) without the need for exemptions. While current NRC guidance continues to assume a LOCA core damage event as the "maximum credible accident" consistent with the historical practice of TID-14844, NRC recognized in 10 CFR 50.67 rulemaking (Reference 8.35) that "there is no regulatory requirement for a specific source term for reactors to be licensed in the future."

In various contexts, NRC staff have addressed source terms for evolutionary and advanced reactor designs where designers have sought to implement the offsite dose evaluation requirements differently due to the nature of the proposed design. For example, in SECY-03-0047 (Reference 8.25), staff recognized that the classic LWR siting evaluation based on an in-vessel core melt may not be applicable to non-LWR designs. Staff recommended the use of "scenario-specific source terms" derived from design basis events defined for the plant, allowing "credit to be given for unique aspects of plant design" (i.e., performance-based). SECY-05-0006 (Reference 8.51) discussed broadening that same framework to include future LWRs.

A.1.2 Standard Review Plan Chapter 15 Off-Site Radiological Consequences

The offsite dose guidelines of 10 CFR 100.11 and 10 CFR 52.47(a)(2)(iv), et al., are the only accident offsite dose limits specified in NRC's regulations. As discussed, that evaluation assesses the consequences of an assumed source term released inside containment, for which the assumed core melt associated with a major LOCA is considered bounding. However, NRC adopted acceptance criteria equal to or derived from those dose consequence guidelines for a spectrum of design basis events analyzed in Chapter 15 of an applicant's Safety Analysis Report. For example, more severe events like a limiting MSLB are compared against the 10 CFR 100.11 criteria, while more likely events such as failure of a small line carrying reactor coolant outside containment can only result in a "small fraction" of those doses.

Accordingly, the MHA is treated as one of the spectrum of design basis radiological events evaluated in Chapter 15 to reasonably assure acceptable offsite doses because the MHA constitutes the design basis of the containment leak rate and the fission product release mitigation systems.

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deleted from the final GDCs, which instead reference a LOCA, reflecting the reliance on ECCS to prevent major core damage.

ACCIDENT SOURCE TERMS REGULATORY FRAMEWORK

A.2 General Design Criterion 19 Control Room Dose

GDC 19 requires a control room "from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions." The radiological dose limits included in GDC 19, 5 rem whole body for the duration of the accident, are intended to provide reasonable assurance of operator safety while performing those functions, such that the operator response is not inhibited. As discussed in Section A.1.1 above, as reactor technology advanced, greater reliance was placed on ESF systems to prevent and mitigate radiological releases from major accidents. In traditional LWRs, operator actions are required to perform those safety functions. Because the core melt MHA constituted the design basis for certain ESFs, it also served as the control room design basis.

A.3 10 CFR 50.49 Environmental Qualification

In the 1970s NRC began reevaluating its requirements for the qualification of electric equipment as required by GDCs 1, 2, 4, and 23. The requirements for EQ of electric equipment in 10 CFR 50.49 were promulgated in 1983, following the TMI accident. By its terms, the scope of 10 CFR 50.49 is primarily "safety-related electric equipment" relied upon for design basis events, including design basis accidents. Some PAM equipment, beyond that which would be considered safety-related, was also considered a "key variable"[13] for assessing the status of the plant. Accordingly, key variables had higher design and quality requirements that may have included qualification in accordance with RG 1.89. For facilities where the core melt MHA serves as the design basis for radiological release mitigation features and control room dose, therefore, the systems relied upon to perform those functions, including the supporting PAM equipment, was required to be qualified to the RG 1.89 source term.

RG 1.89, which addresses EQ, assumes a radiological source term for a "design basis LOCA" equivalent to a core melt MHA: 100% of the noble gas activity, 50% of the halogen activity, and 1% of the remaining fission product activity. Several of the TMI requirements address monitoring capability following severe accidents, which typically relies on PAM equipment. Accordingly, for facilities where a core melt MHA serves as the design basis for certain SSCs there is overlap between electric equipment relied upon for design basis events and that relied upon for severe accidents. In the case of PAM equipment, NRC developed a revision of RG 1.97 (Reference 8.56) to address the several monitoring requirements in a cohesive manner to satisfy the various requirements involved. Because of NuScale's use of the Iodine Spike MHA and Category 1 events for EQ, there is less overlap between 10 CFR 50.49 EQ for PAM equipment and TMI severe accident design provisions.

[13] A "key variable … most directly indicates the accomplishment of a safety function … or the operation of a safety system … or radioactive material release. It is essential that key variables be qualified to the more stringent design and qualification criteria." (Reference 8.56)

A.4 Severe Accident Mitigation

A.4.1 TMI requirements and Equipment Survivability

The TMI Action Plan included a series of requirements broadly aimed at improving "the capability of nuclear power plants to mitigate the consequences of accidents in which the core is severely damaged" (Reference 8.36). NRC also directed actions to improve radiation protection of workers and the public under accident conditions.

The requirements of 10 CFR 50.34(f)(2)(vii), (viii), (xxvi), and (xxviii) require that an applicant address "source term radioactive materials" following an accident for the purposes of shielding design, sampling capability, leakage control and detection of systems outside containment, and control room leakage pathways, respectively. The requirements include reference to the source term footnote originally from 10 CFR 100.11; i.e., "a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible." The well-established core melt source term of RGs 1.3 and 1.4 (based on TID-14844) served as a surrogate degraded core event for demonstrating these facility capabilities.

The Severe Accident Policy Statement (Reference 8.37) provided that new power plant designs must be shown to be acceptable for severe accident concerns by, amongst other criteria, demonstration of compliance with the TMI requirements of 10 CFR 50.34(f)[14] and consideration of severe accident vulnerabilities exposed by a PRA. In SECY-90-016, staff addressed the issue of assuring that severe accident mitigation features are demonstrated to be available to perform their functions. Staff concluded that severe accident "mitigation features must be designed so there is reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed." Because "severe core damage accidents should [not] be design basis accidents (DBA) in the traditional sense that DBAs have been treated in the past," in that features provided for only severe accident mitigation were not required to meet typical safety-related requirements.[15]

The monitoring requirements of 10 CFR 50.34(f)(2)(xvii) through (xix) do not specify a particular source term, but they each address various aspects of monitoring plant status during and following an accident that may lead to or involve core damage. The latter of these rules, 10 CFR 50.34(f)(2)(xix), explicitly requires that an applicant "provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage." For operating reactors, these and other monitoring requirements are addressed collectively by the spectrum of monitoring variables included in RG 1.97.

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[14] At that time, the TMI requirements were only applicable to construction permit applications under Part 50; they were later explicitly adopted as Part 52 application requirements.

[15] "Features…need not be subject to (a) the 10 CFR 50.49 environmental qualification requirements, (b) all aspects of 10 CFR Part 50, Appendix B quality assurance requirements, or (c) 10 CFR Part 50, Appendix A redundancy/diversity requirements." (Reference 8.24)

For NuScale, these monitoring requirements are met by a separately considered list of equipment because of the lack of a core damage design basis event.

A.5 Emergency Planning

Although emergency planning is outside the scope of the NuScale DCA, it will be addressed by applicants referencing the NuScale design. As the last layer of defense-in-depth, severe accident source terms have historically been addressed by emergency planning.

A.5.1 Emergency Planning Zone

As discussed in SECY-15-0077 (Reference 8.39), the current regulations of 10 CFR 50.47(c)(2) and Appendix E rules reflect presumed EPZ sizes of 10 miles and 50 miles for the plume exposure and ingestion exposure pathways, respectively. The technical basis for the determination of the EPZ size considered various accident sequences for large LWRs and recommended the EPZ sizes by weighing the dose savings to the public against the likelihood of the various sequences. In this manner, the standard EPZ sizes are loosely derived from hypothetical source terms for traditional large LWRs. Industry and NRC are currently pursuing new methodologies to establish SMR EPZ sizes commensurate with the offsite radiological risk of a particular design, which would more closely couple the EPZ sizes to postulated design-specific source terms. NuScale's proposed approach is described in Reference 8.17 and summarized in Section 6.2.5 of this paper.

A.5.2 Emergency Response Monitoring

Instrumentation that supports an emergency response includes considerations of source terms to provide reasonable assurance of SSC performance. For example, 10 CFR 50 Appendix E IV.B.1 requires that the Emergency Plan include a description of "means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials." That means is provided by the Type E variables pursuant to RG 1.97. Likewise, NUREG-0696 requires that the TSC and EOF include data from the Type A, B, C, D, and E variables addressed in RG 1.97.

RG 1.89 normally bases radiological EQ of that instrumentation on a core melt MHA. NuScale's use of the Iodine Spike MHA leads to use of the separately considered equipment associated with 10 CFR 50.34(f)(2)(xix) to achieve the monitoring required to coordinate emergency responses.

A.5.3 Emergency Response Facilities

10 CFR 50 Appendix E IV.E requires the licensee to provide an onsite TSC. NUREG-0696 requires that the TSC "have the same radiological habitability as the control room under accident conditions," with relaxed quality and design requirements for habitability systems. Accordingly, TSC personnel dose is normally evaluated for the core melt MHA source term. NuScale uses the non-core damage Iodine Spike MHA and Category 1 events for this purpose.

ACCIDENT SOURCE TERMS REGULATORY FRAMEWORK

Appendix B. Overall Source Term Framework

Source Term Name(s)	Design Basis Uses of Source Term(s)
Accident Source Terms	
Category 1 (Main steam line break, failure of small lines carrying primary coolant outside containment, rod ejection accident, steam generator tube failure, and fuel handling accident) source terms	• SRP 15.0.3 dose consequences • EQ (accident dose) • Post-accident operator access • Control room habitability • TSC habitability
Category 2 (MHA) source term	• 10 CFR 52.47(a)(2)(iv) dose consequences • EQ (accident dose) • Post-accident operator access • Control room habitability • TSC habitability
Radioactive Waste Systems and Effluent Related Source Terms	
Normal realistic, effluent release, and normal effluent source terms	• Normal effluents offsite dose consequences • 10 CFR 50 Appendix I
Design basis shielding, dose, and liquid radioactive waste tank failure source terms	• Radiation shielding • Radioactive waste system design • HVAC design • EQ (normal dose) • 10 CFR 20, Appendix B • BTP 11-6 and GDCs 60, 61, and 64
Gaseous radioactive waste system failure source term	• Gaseous radioactive waste system design • BTP 11-5 and GDC 61
Beyond Design Basis Core Damage Source Terms	
Equipment survivability source term[16]	• 10 CFR 50.34(f)(2)(xix) • Design capability to collect sufficient data to inform emergency plans.
SAMDA source terms[17]	• 10 CFR 51.55 • Establish the risk associated with severe accidents and perform a cost-benefit analysis to demonstrate bases for not incorporating SAMDAs into the design to be certified.
EPZ source term[18]	• 10 CFR 50.47 and 10 CFR 50 Appendix E • Establish the size of the EPZ

[16] NuScale plans to assume a radiological release from a core damage event to inform a qualitative assessment of Equipment Survivability in FSAR Section 19.2.3.3.8.
[17] SAMDA source terms do not have a design basis purpose; however they are still shown for context.
[18] The EPZ source term is outside of the scope of the NuScale DCA, however it is shown for context.