RAIO-0118-58186

January 15, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 71 (eRAI No. 8889) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 71 (eRAI No. 8889)," dated June 27, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 71 (eRAI No.8889)," dated August 16, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) supplemental response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's supplemental response to the following RAI Question from NRC eRAI No. 8889:

- 19-13

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8889

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8889

NuScale Power

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8889
Date of RAI Issue: 06/27/2017

NRC Question No.: 19-13

Regulatory Basis

10 CFR 52.47(a)(27) states that a design certification application (DCA) must contain a Final Safety Analysis Report (FSAR) that includes a description of the design-specific probabilistic risk analysis (PRA) and its results. 10 CFR 52.47(a)(2) states that the standard plant should reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of radioactive fission products. 10 CFR 52.47(a)(4) states that each DCA must contain an FSAR that includes and analysis and evaluation of the design and performance of systems, structure and components (SSCs) with the objective of assessing the risk to public health and safety resulting from operation of the facility and including a determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents.

Request for additional information

NuScale FSAR Chapter 19, page 19.2-27 states "The probability of [a steam generator tube failure (SGTF)] during high-temperature severe accident conditions was developed conservatively assuming the primary side was depressurized and the secondary side was pressurized. The probability of such a failure is incorporated into the Level 2 PRA as described in Section 19.1.4.2."

The applicant's analysis of the probability of SGTF is described in FSAR Section 6.4.1 "Thermal-Induced Steam Generator Tube Failure" of ER_P020_7024_R0, "Level 2 Probabilistic Risk Assessment Notebook." The applicant's analysis uses the methodology described in Section 2.5 "SGTF Probability under Severe Accident Conditions" of ER_P010_3782_R0 "Steam Generator Tube Failure Probabilistic Risk Assessment Report."

For severe accident scenarios in pressurized water reactors (PWRs) that feature U-tube steam generators with the primary system at high pressure and a dry secondary system at low pressure (known as the "high-dry-low" scenario), a counter-current flow of hydrogen and superheated steam occurs in the hot legs and in the steam generator tubes. This phenomenon was demonstrated in the Westinghouse 1/7th scale experiments. In the high-dry-low scenario, ex-vessel piping is exposed to high temperatures and high internal pressures. For the high-dry-low scenario, studies have been performed to estimate the probability of a steam generator tube rupture before another ex-vessel piping rupture (which could result in higher offsite radiological consequences if a tube ruptures).

The applicant extrapolated the results of one of these studies (Liao, Y., and Guentay, S., “Potential steam generator tube rupture in the presence of severe accident thermal challenge and tube flaws due to foreign object wear,” Nuclear Engineering and Design, Vol. 239, Issue 6, pp: 1128-1135, 2009) to estimate the probability of thermally induced steam generator tube rupture for NuScale. However, the applicability of the Liao and Guentay study to NuScale is unclear because of design differences. For example, the probability distribution developed by Liao and Guentay represents the probability of a steam generator tube rupturing before other ex-vessel piping (hot leg and surge line) ruptures. NuScale does not have ex- vessel piping. Also, the design analyzed by Liao and Guentay has primary coolant on the inside of the steam generator tubes. NuScale has primary coolant on the outside of the steam generator tubes. The applicant is requested to justify the applicability of these studies to NuScale's design.

NuScale Response:

NuScale provided its original response to RAI 8889, Question 19-13, in letter RAIO-0817-55461, dated August 16, 2017. Per discussion with the staff during a public meeting on October 03, 2017, NuScale is revising its original response regarding evaluation of the steam generator tube failure (SGTF) probability. The response is revised to be consistent with a revision to the PRA that reflects an updated analysis of the thermally induced SGTF probability during a severe accident. Specifically, the updated analysis evaluates the probability of a thermally induced SGTF during a severe accident without use of the probability distribution developed by Liao and Guentay for a steam generator tube rupture before other ex-vessel piping ruptures (e.g., hot leg or surge line). Accordingly, the following response replaces the original response, in its entirety:

As discussed in FSAR Section 19.1.4.1.1.5, the operating characteristics of the NuScale steam generators are opposite of those in conventional pressurized water reactors (PWRs); notably, secondary coolant flows through the inside of the steam generator tubes and the higher-pressure primary coolant is on the outside such that the tubes are maintained in a constant state of compression. As such, NuScale uses the term steam generator tube "failure" to distinguish potential tube faults from the steam generator tube “rupture” (SGTR) terminology that is used in conventional PWRs. The NuScale steam generator is also helical by comparison to a U-tube or once-through design. Along with the use of Alloy-690, these design differences are expected to result in less frequent tube failures. Those failures are expected to be less severe (i.e., the leak area of a crushed or buckled tube is significantly less than a ruptured tube).

In the NuScale design, reactor coolant flow is driven by natural circulation; reactor coolant flows upward through the core due to natural circulation where it absorbs heat and continues to flow up the central riser. At the top of the central riser, it is turned by the pressurizer baffle plate and flows downward across the integrated steam generator tube bundles. Heat is removed by helical steam generator tubes coiled inside the reactor pressure vessel at a distance above the reactor core. Feedwater enters the steam generators through the feedwater supply nozzles and feed plenums and flows up the helical tubes where it is heated, boiled, and superheated before it exits through the steam plenums and main steam supply nozzles to the steam lines.

As discussed in FSAR Section 19.2.3.3.6, the high temperature and pressure differentials across the steam generator tubes during a severe accident can induce an SGTF. Under severe accident conditions, tubes have a higher failure probability than during normal operation due to high temperature creep failure. To estimate the probability of an SGTF during a severe accident in the NuScale design, a high temperature creep failure evaluation has been performed based on historical data for tube flaws experienced in conventional steam generators and consideration of the temperature and pressure conditions to which the tubes would be exposed during a postulated NuScale severe accident. The formulations employed for predicting creep rupture are based on internally pressurized tubes; however, the NuScale steam generator tubes are externally pressurized. As a result, the calculated probability of a thermally induced SGTF is judged to be overestimated because creep progresses more vigorously under tension than compression.

Creep failure is evaluated using a concept that the integral "life left" of a steam generator tube is degraded over time as a function of time-history temperature and stress conditions. The tube failure/creep rupture model is employed with the mean-time-to-rupture correlations from NUREG-1570. The nominal temperature and stress conditions that the tubes are exposed to are derived from a MELCOR simulation of a NuScale Power Module (NPM) severe accident for 30 days. Uncertainty is accounted for in the temperature and pressure conditions, as well as the Larson-Miller parameter, by imposing a distribution about the nominal values. A Monte Carlo technique that employs two steps is used to converge on a mean probability of a thermally induced SGTF. In the first step, probability distributions of steam generator tube flaw frequency and flaw depths are correlated to a stress magnification factor probability of exceedance. In the second step, randomly sampled severe accident temperature and stress conditions are used to solve for the critical stress magnification factor (i.e., the minimum stress magnification factor that would result in creep failure). The probability that the stress magnification factor exceeds the critical stress magnification factor is recorded for each set of randomly sampled parameters to provide a probability distribution of a thermally induced SGTF. The resulting mean probability for a thermally induced SGTF during a severe accident is 2.5E-02 per steam generator.

The probability of a thermally induced SGTF has been updated in FSAR Table 19.1-10. This change has a negligible effect on the large release frequency (LRF), as illustrated in the revised LRF values summarized in FSAR Sections 19.1.4.2.2, 19.1.9.1 and Table 19.1-80. The format of Figure 19.1-14 has been simplified and reflects a negligible change in the LRF. There is no change in the core damage frequency; however, the format of Figure 19.1-13 has been modified for consistency with Figure 19.1-14. The change in the probability of a thermally induced SGTF does not introduce new candidates for risk significance.

FSAR Tables 19.1-22 and 19.1-31 have been updated to separate the SGTF initiating event frequency sensitivity from the severe accident induced SGTF probability sensitivity. Industry data were used in the SGTF initiating event sensitivity. The results show negligible changes in core damage frequency (CDF) and LRF. Thermally induced SGTFs that result in radiological consequences have a very low probability in the NuScale design because failure of at least two isolation valves is also required for a release. As shown by Sensitivity Study 9, summarized in FSAR Table 19.1-31, increasing the probability of a severe accident thermally induced SGTF using the 95th percentile value of the creep rupture model results in negligible changes to LRF. Additional clarifications have been made to the sensitivity study results for consistency with the changes to the thermally induced SGTF modeling.

The key assumption related to thermally induced SGTF that is provided in FSAR Table 19.1-28 has been revised to reflect the use of a creep rupture model that is based on historical data and conditions representative of a NuScale severe accident.

Impact on DCA:

FSAR Sections 19.1.4.2.2, 19.1.9.1, 19.2.3.3.6; FSAR Tables 19.1-10, 19.1-22, 19.1-28, 19.1-31, 19.1-80; and FSAR Figures 19.1-13 and 19.1-14 have been revised as described in the response above and as shown in the markup provided in this response.

accident progression modeling is performed with NRELAP5 and MELCOR as described in Section 19.1.4.1.1.6.

19.1.4.2.1.7 Quantification

Linking of the Level 2 CET and system models to quantify the Level 2 results is performed using the SAPHIRE software in the same manner as is performed in the Level 1 analysis. By physically linking the Level 1 system models with the Level 2 system models, system dependencies are explicitly captured.

An appropriate truncation level ensures that dependencies and significant accident sequences are not eliminated from the evaluation. Consistent with the ASME/ANS PRA Standard, a convergence analysis was performed with SAPHIRE to evaluate the point at which less than a five percent change in LRF occurs when the truncation level is reduced by a factor of ten. Based on the convergence results, a truncation value of 1E-15 is used for the LRF.

19.1.4.2.1.8 Uncertainty

The types and treatment of uncertainty associated with the Level 2 PRA are the same as discussed in Section 19.1.4.1.1.8 for the Level 1 PRA.

19.1.4.2.2 Results from the Level 2 Probabilistic Risk Assessment for Operations at Power

This section provides results of the Level 2 PRA for full power operation of a single module. Large release frequency and insights on the significant contributors to the calculated large release frequency are presented. Uncertainties and sensitivity studies associated with the results are discussed.

<u>Large Release Frequency</u>

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The mean value of the large release frequency for a single module due to internal events at-power for a module was calculated to be 2.~~1~~<u>3</u>E-11 per mcyr; the 5th and 95th percentile values are 1.~~2~~<u>4</u>E-13/mcyr and 5.~~7~~<u>8</u>E-11/mcyr, respectively.

<u>Conditional Containment Failure Probability</u>

The conditional containment failure probability (CCFP), defined as the ratio of LRF to CDF, for full-power operation of a module is 0.06 for the internal events core damage sequences assessed in the PRA.

<u>Significant Large Release Sequences</u>

The significant core damage sequences that contribute to the LRF are provided in Table 19.1-26. The table provides the sequence identifier, the percentage contribution to the LRF, and a summary description of the sequence. The table illustrates that the large release frequency is dominated by sequences with a CVCS

- conformance with safety goals
- perspective of the NuScale small core with respect to safety goals
- focused PRA insights
- unique system capability

19.1.9.1 Conformance with Safety Goals

The safety goal policy statement and subsequent guidance provide quantitative objectives for evaluating conformance with the qualitative goals associated with public health and safety. The quantitative results of the PRA, summarized in Table 19.1-82, demonstrate that the risk associated with operation of a NuScale Power Module is substantially less than defined by the safety goals. The table also indicates that additional risk associated with multiple module operation is small. As indicated in the table:

- the mean value of the CDF of a NuScale module is 3.0E-10/mcyr as compared to the CDF safety goal of 1.0 E-4 per reactor year.
  - The ATWS contribution to CDF is 2.2E-11/mcyr, significantly less than the target of 1.0E-5 per reactor year provided in SECY 83-293.
  - With regard to a multi-module configuration, the MM-CDF is about 10 percent of the CDF.

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- the mean value large release frequency of a NuScale module, LRF, is 2.~~1~~3E-11/mcyr as compared to the LRF safety goal of 1.0 E-6 per reactor year.
  - With regard to a multi-module configuration, the MM-LRF is less than 10 percent of the LRF.
- the composite CCFP of a module is less than the safety goal of 0.1.
- the evaluated external events (seismic, internal fire, internal flood, external flood, and high winds) do not pose a significant risk to the plant.

RAI 19-27

The CDF and LRF risk metrics illustrate conformance with the quantitative health objectives (QHOs) defined in Reference 19.1-36. Conformance with the prompt fatality QHO is illustrated by an LRF that is well below the surrogate risk metric of less than 1×10^{-6} per reactor year. Similarly, risk results show that NuScale demonstrates conformance with the latent cancer QHO as illustrated by a CDF that is well below the surrogate metric of less than 1×10^{-4} per reactor year.

RAI 19-31

COL Item 19.1-8: A COL applicant that references the NuScale Power Plant design certification will confirm the ~~applicability~~validity of assumptions and data used in the design certification application and modify, as necessary, for applicability to the ~~to the~~ as-built/as-operated probabilistic risk assessment.

19.1-61 NUREG/CR-5497 - 2012 Update, “Common Cause Failure Parameter Estimations,” U.S. Nuclear Regulatory Commission, January 2012.

19.1-62 Microsemi Reliability Report, No. 51000001-11/05.13, May 2013.

RAI 19-27

19.1-63 SAND2011-0128, "Accident Source Terms for Light Water Nuclear Power Plants Using High-Burnup or MOX Fuel," Sandia National Laboratories, January 2011.

RAI 19-34

19.1-64 NUREG-1524, “A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of the Current Understanding of Broader Fuel-Coolant Interaction Issues,” U.S. Nuclear Regulatory Commission, July 1996.

RAI 19-13S1

19.1-65 NUREG-1570, “Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture,” March 1998.

RAI 19-13S1

Table 19.1-10: Basic Events Requiring Design-Specific Analysis

Description	Mean	Uncertainty	Use
CVCS LOCA does not initiate excess flow check valve	1E-01	EF = 10; lognormal	For sequences when there is a potential that the flow rate of a LOCA, resulting from a leak or small break, does not engage the CVCS excess flow check valve to isolate, this failure probability of the valve to close is assumed based on engineering judgment.
ECCS reactor vent valve passive opening at low differential pressure	1E-01	EF = 10; lognormal	When the dp across the valve gets low for a sufficient amount of time, the spring force becomes the dominant term in the force balance and pulls the main valve open. This characteristic of passive opening is considered when a valve fails to open on demand; the failure probability to open passively is assumed based on engineering judgment.
ECCS reactor recirculation valve passive opening at low differential pressure	1E-01	EF = 10; lognormal	When the dp across the valve gets low for a sufficient amount of time, the spring force becomes the dominant term in the force balance and pulls the main valve open. This characteristic of passive opening is considered when a valve fails to open on demand; the failure probability to open passively is assumed based on engineering judgment.
Probability that the RSV is demanded to open	5E-01	N/A	In sequences when there is a small pressure margin for an RSV demand, the probability that an RSV is demanded to open is considered; this probability is based on engineering judgment.
DHRS train passive heat transfer to reactor pool	4E-06	EF = 2; lognormal	Following successful actuation of a DHRS train, this event represents a failure of passive heat transfer (i.e., natural circulation) to the UHS over the mission time.
ECCS passive heat transfer to reactor pool	1E-07	EF = 3; lognormal	Following successful actuation of ECCS, this event represents a failure of passive heat transfer (i.e., natural circulation) to the UHS over the mission time.
MPS test or maintenance unavailability for: • Scheduling and bypass module • SFM • Scheduling and voting module SVM)	 2.7E-03 2.7E-03 2.1E-03	 EF = 10; lognormal EF = 10; lognormal EF = 10; lognormal	Twelve hours of maintenance between refueling outages (i.e., 2 years) is assumed to result in unavailability of each MPS channel. When an MPS channel is in maintenance, it is placed into trip or bypass.
Temperature induced SGTF	~~5.6E-04~~2.5E-02	EF = ~~10~~2; lognormal	The conditional probability that a helical coil steam generator tube (in compression) fails following core damage.

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Table 19.1-22: Sensitivity Studies for Level 1 Full Power, Internal Events Evaluation

Item	Modeling Assumption or Uncertainty	Sensitivity Study	Basis	Result
1.	Human Error Probability	Effect of HEP	Generic uncertainty identified in EPRI TR-1016737	Safety goals met irrespective of selected HEP.
1a.	5th percentile value	All HEPs (pre-initiator, post initiator and dependent probabilities) set to guaranteed success.	Bounding representation of 5^{th} percentile value	CDF decreased ~~by 27%~~ slightly.
1b.	95th percentile value	All HEPs (pre-initiator, post initiator and dependent probabilities) set to guaranteed failure.	Bounding representation of 95^{th} percentile value	CDF increased by two orders of magnitude in comparison to base case.
2.	Common cause failure	All CCFs set to 0.002, the mean value of the highest CCF demand event	Generic uncertainty identified in EPRI TR-1016737. Modeling simplification judged representative of 95th percentile value.	CDF increases to ~~4.2E-06~~4E-06/ mcyr
3.	Initiating Event Frequency	Effect of uncertainty in initiating event frequencies.	Initiating event frequency for some initiating events is site-specific and/or dependent on generic data	Safety goals met irrespective of initiating event frequency
3a.	IE-EHVS-LOOP	LOOP frequency increased to 1.0 per year (from 3.1E-02). LOOP frequency decreased by an order of magnitude (to 3.1E-3).	Address uncertainty of grid stability at various sites	CDF increase by a factor of 8~~3~~. CDF decrease~~ by a factor of 0.8.~~d slightly.
3b.	IE-MSS---ALOCA-SG-	SGTF frequency increased from 4.0E-05 to 1.4E-03 per year~~, and probability of induced SGTF increased from 5.6E-04 to 2.2E-01~~.	Address uncertainty in unique design feature; value based on 2010 industry average data.	Negligible CDF change.
3c.	IE-TGS---FMSLB-UD	Secondary line break initiating event frequency increased from 4.4E-05 to 7.7E-03	Address uncertainty in nonsafety related initiator frequency; value based on 2010 industry average data.	~~10% increase in CDF.~~CDF increased slightly.
3d.	IE-CVCS---ALOCA-CIC	Increase CVCS injection line LOCA inside containment initiating event frequency by an order of magnitude	Address design-specific uncertainty of very small LOCA inside containment resulting in containment isolation.	~~20% increase in CDF.~~CDF increased slightly.
4.	Passive heat removal reliability	Increase the failure probability of passive heat removal (ECCS, DHRS) by an order of magnitude	Address the design-specific uncertainty of passive decay heat removal, including UHS reliability.	~~20% increase in CDF.~~CDF increased slightly.
5.	ECCS opening on low differential pressure	Increase the failure probability of ECCS opening on low differential pressure from 0.1 to 0.5.	Address the design-specific uncertainty of ECCS success criteria	CDF increase by a factor of 2~~.3~~.
6.	Failure probability of sensors	Increase the failure probability of sensors an order of magnitude	Address the design-specific uncertainty of potentially utilizing new technologies (e.g., digital components) to monitor plant parameters.	Negligible CDF change.

Table 19.1-22: Sensitivity Studies for Level 1 Full Power, Internal Events Evaluation (Continued)

Item	Modeling Assumption or Uncertainty	Sensitivity Study	Basis	Result
7.	Credit for non-safety systems	Focused PRA which credits only safety-related systems performed to evaluate RTNSS Criterion C.	Evaluate effect of crediting only safety-related systems with regard to safety goal conformance.	Safety goals met without credit for nonsafety-related systems: CDF is ~~3.0E-6~~3E-06/mcyr
8.	Core damage system importance and the use of generic data	Evaluate system importance for core damage; PRA systems are identified in Table 19.1-4	Evaluate PRA system importance to address the uncertainty of using generic LWR component failure data due to the absence of design-specific operating experience; evaluate system importance against the core damage risk significance threshold of CCDF $\geq 1 \times 10^{-5}$ /year, as identified in Table 19.1-20.	Three systems meet the core damage threshold for risk significance: ECCS, MPS, and the UHS. These systems are safety-related. In addition, the MPS comprises both the RTS and ESFAS subsystems.

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Table 19.1-28: Key Assumptions for the Level 2 Full Power Internal Events Probabilistic Risk Assessment

Assumption	Basis
Containment penetrations are screened if they are sealed, normally closed, or formed a closed loop inside containment; screened penetrations are assumed to be negligible contributors to the potential for a containment release. The CNV is maintained at a vacuum during normal operation.	Common engineering practice and engineering judgment.
Only the first two CIVs are modeled; many lines include additional isolation valves that are not considered.	Bounding simplification
A single sensor group is modeled to initiate containment isolation. The design includes multiple sensor groups that may initiate containment isolation.	Bounding assumption
The probability of a thermally induced SGTF is based on a ~~probabilistic physics of failure approach~~creep rupture model that uses historical data for conventional SG tube flaws and time-history temperature and pressure conditions representative of NuScale severe accident progression. In the NuScale design, the steam generator tubes are in compression (i.e., feedwater is on the inside and primary reactor coolant circulates on the outside) which is opposite the typical tensile stresses in conventional plants. A thermally induced SGTF is assumed to result in a double-ended rupture of a single tube.	Engineering analysis and judgment
Core damage sequences are binned into a single plant damage state (i.e., core damage). Source term release categories are binned into two release categories (i.e., core damage with containment isolation, and core damage with failure of containment isolation or bypass). Additional plant damage states and release categories are not needed to support evaluation of a large release.	Common engineering practice and bounding assumption
Core damage sequences that include containment bypass or failure of containment isolation are typically assumed to result in a large release; a large release is defined as a release that results in an acute whole body 200 rem dose to a hypothetical individual located at the reactor site boundary over the course of 96 hours.	Bounding assumption
Mitigating factors such as fission product deposition, retention or scrubbing of fission products (e.g., spray or filtration), or deflection or absorption (i.e., biological shield) are not credited in the Level 2 PRA.	Bounding assumption
The Level 2 PRA does not credit recovery of the containment envelope (e.g., through implementation of severe accident management guidelines), if lost.	Bounding assumption

RAI 19-13S1, RAI 19-25

Table 19.1-31: Sensitivity Studies for Level 2 Evaluation

Item	Modeling Assumption or Uncertainty	Sensitivity Study	Basis	Result
1.	Human Error Probability	Effect of HEP	Generic uncertainty identified in Reference 19.1-7 (EPRI TR-1016737)	Safety goals met irrespective of selected HEP
1a.	5th percentile value	All HEPs (pre-initiator, post initiator and dependent probabilities) set to guaranteed success.	Bounding representation of 5th percentile value	LRF decreased slightly.
1b.	95th percentile value	All HEPs (pre-initiator, post initiator and dependent probabilities) set to guaranteed failure.	Bounding representation of 95th percentile value	LRF increased by over 2 orders of magnitude in comparison to base case.
2.	Common cause failure	All CCFs set to 0.002, the mean value of the highest CCF demand event.	Generic uncertainty identified in EPRI TR-1016737. Modeling simplification judged representative of 95th percentile value.	LRF increases to ~~3.7E-08~~4E-08/mcyr.
3.	Initiating Event Frequency	Effect of uncertainty in initiating event frequencies.	Initiating event frequency for some initiating events is site-specific and/or dependent on generic data	Safety goals met irrespective of initiating event frequency.
3a.	IE-EHVS-LOOP	LOOP frequency increased to 1.0 per year (from 3.1E-02). LOOP frequency also decreased one order of magnitude.	Address uncertainty of grid stability at various sites	Negligible LRF change
3b.	IE-MSS---ALOCA-SG-	SGTF frequency increased from 4.0E-05 to 1.4E-03 per year~~, and probability of induced SGTF increased from 5.6E-04 to 2.2E-01~~.	Address uncertainty in unique design feature; value based on 2010 industry average data.	~~LRF increase by less than a factor of 2~~Negligible LRF change
3c.	IE-TGS---FMSLB-UD	Secondary line break initiating event frequency increased from 4.4E-05 to 7.7E-03.	Address uncertainty in nonsafety related initiator frequency; value based on 2010 industry average data.	Negligible LRF change
3d.	IE-CVCS—ALOCA-CIC	Increase CVCS injection line LOCA inside containment initiating event frequency by an order of magnitude	Address design-specific uncertainty of very small LOCA inside containment resulting in containment isolation.	Negligible LRF change
4.	Passive heat removal reliability	Increase the failure probability of passive heat removal (ECCS, DHRS) by an order of magnitude	Address the design-specific uncertainty of passive decay heat removal, including UHS reliability.	Negligible LRF change
5.	ECCS opening on low differential pressure	Increase the failure probability of ECCS opening on low differential pressure from 0.1 to 0.5.	Address the design-specific uncertainty of ECCS success criteria.	Negligible LRF change

Table 19.1-31: Sensitivity Studies for Level 2 Evaluation (Continued)

Item	Modeling Assumption or Uncertainty	Sensitivity Study	Basis	Result
6.	Failure probability of sensors	Increase the failure probability of sensors an order of magnitude.	Address the design-specific uncertainty of potentially utilizing new technologies (e.g., digital components) to monitor plant parameters.	Negligible LRF change
7.	Credit for non-safety systems	Focused PRA which credits only safety-related systems performed to evaluate RTNSS Criterion C.	Evaluate effect of crediting only safety-related systems with regard to safety goal conformance.	Safety goals met without credit for nonsafety-related systems: LRF is ~~1.5E-7~~2E-07/mcyr
8.	Large release system importance and the use of generic data	Evaluate system importance for large release; PRA systems are identified in Table 19.1-4.	Evaluate PRA system importance to address the uncertainty of using generic LWR component failure data due to the absence of design-specific operating experience; evaluate system importance against the large release risk significance threshold of CLRF $\geq 1 \times 10^{-6}$ /year, as identified in Table 19.1-20.	Two systems meet the large release threshold for risk significance: CNTS and MPS. These systems are safety-related.
9.	Probability of induced SGTF	Increase the probability of an induced SGTF using the 95th percentile value of the creep rupture model.	Address the uncertainty in creep rupture failure on SG tubes.	Negligible LRF change

RAI 19-13S1

Table 19.1-80: Summary of Results (Mean Values)

Full Power			
Hazard	**CDF (per mcyr)**	**LRF (per mcyr)**	**Applicable DCD Section**
Internal Events	3.0E-10	2.~~1~~3E-11	Section 19.1.4.1.2 and Section 19.1.4.2.2
Internal Fires	9.7E-10	4.3E-11	Section 19.1.5.2.2
Internal Floods	6.1E-11	<1E-15	Section 19.1.5.3.2
External Floods	1.3E-09	1.0E-13	Section 19.1.5.4.2
High Winds (Tornado)	1.5E-10	<1E-15	Section 19.1.5.5.2
High Winds (Hurricane)	1.0E-09	6.6E-14	Section 19.1.5.5.2
Seismic (SMA)	0.88g[1]	0.88g[1]	Section 19.1.5.1.2
Low Power and Shutdown			
Hazard	**CDF (per year)**	**LRF (per year)**	**Applicable DCD Section**
Internal Events	4.5E-13	2.3E-14	Section 19.1.6.2
Module Drop	8.8E-08	N/A[2]	Section 19.1.6.2
Internal Fires	Negligible	Negligible	Section 19.1.6.3.2
Internal Floods	Negligible	Negligible	Section 19.1.6.3.3
External Floods	Negligible	Negligible	Section 19.1.6.3.4
High Winds (Tornado)	1.4E-13[4]	<1E-15[4]	Section 19.1.6.3.5
High Winds (Hurricane)	2.3E-11[4]	2.5E-15[4]	Section 19.1.6.3.5
Seismic (SMA)	Negligible	Negligible	Section 19.1.6.3.1
Multi-Module			
Hazard	**Conditional Probability of Core Damage**	**Conditional Probability of Large Release**	**Applicable DCD Section**
Multi-Module Factor Reduction	0.13[3]	0.01[3]	Section 19.1.7.2
Composite CCFP < 0.1			

Notes**:**

[1] A seismic margins assessment was performed; results are presented in terms of the HCLPF i.e., peak ground acceleration at which there is 95% confidence that the conditional failure probability is less than 5%. Note that these results are driven by the bounding assumption that a structural failure results in both core damage and a large release.

[2] A module drop does not result in a large release.

[3] Results are presented in terms of a bounding estimate on the conditional probability that multiple modules would experience core damage (or large release) following core damage (or large release) in a single module.

[4] Results are point estimates.

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Figure 19.1-13: Contribution to Internal Events Core Damage Frequency by Initiator ~~(Point Estimates)~~

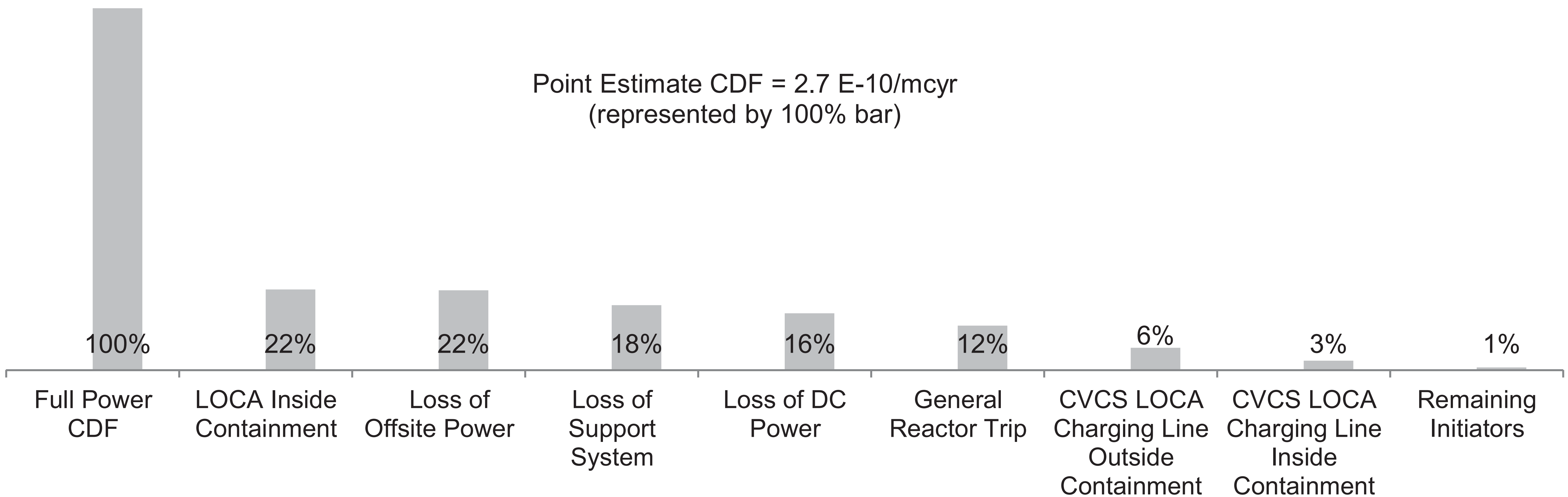

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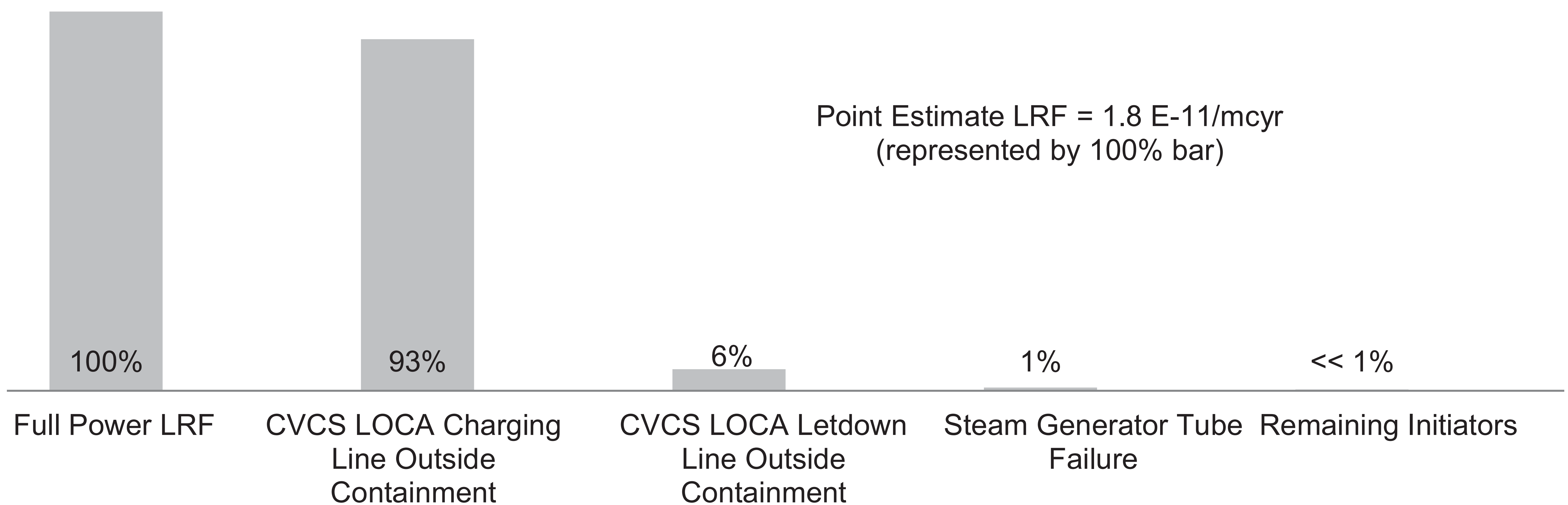
Figure 19.1-14: Contribution to Internal Events Large Release Frequency by Initiator ~~(Point Estimates)~~



19.2.3.3.6 Containment Bypass

A containment bypass is a flow path that allows an unintended release of radioactive material directly to the Reactor Building, bypassing containment. Core damage sequences that include containment bypass or failure of containment isolation are assumed to result in a large release as defined in Section 19.1.4.2.1.4. No distinction is made between "early" or "late" releases. Containment bypass could occur through (i) failure of containment isolation or (ii) steam generator tube failure (SGTF) concurrent with failure of secondary-side isolation on the failed steam generator (SG). Containment bypass is represented by top event CNTS-T01 as discussed in Section 19.1.4.2.1.3.

Containment Isolation Failure

As stated in Section 6.2.4, the containment system design provides for isolation of systems that penetrate the CNV. The design is reflected in a containment isolation and bypass model as summarized in Section 19.1.4.2.1.3.

Thermally-Induced Steam Generator Tube Failure

In the NuScale design, the SG bundles are integrated within the RPV; they form part of the RPV reactor coolant pressure boundary. In contrast with conventional pressurized water reactors, the primary reactor coolant circulates over the outside of the SG tubes, with the steam-formation occurring in the secondary coolant on the inside of the SG tubes. As such, the NuScale SG tubes operate with the higher primary pressure on the outside of the tubes and lower secondary pressure on the inside of the tubes. The result is that there are predominately compressive stresses on the tubes versus the typical tensile stresses. Because the mechanism for fatigue crack propagation is tensile stress, the NuScale SG pressure conditions are expected to prevent crack propagation.

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Due to the lack of data on thermal-induced SGTFs for the NuScale design, ~~a probabilistic physics of failure approach was used to simulate damage accumulation (due to crack and wear growth) during operation using physical models of damage.~~an evaluation of creep rupture was performed based on historical data for conventional SG tube flaws and time-history temperature and pressure conditions representative of NuScale severe accident sequences.

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The SG tubes under severe accident conditions typically have a much higher probability of failure because of the higher temperatures during a severe accident. ~~A creep rupture model for predicting SGTF, informed by tests on unflawed and flawed specimens containing axial and circumferential flaws and stressed under constant as well as ramped temperature and pressure conditions, was used to predict the failure temperature and time.~~ The probability of an SGTF ~~during high-temperature severe accident conditions was developed conservatively assuming the primary side was depressurized and the secondary side was pressurized~~is calculated using the tube failure /creep rupture model presented in NUREG-1570

(Reference 19.1-65). Although the formulations employed for predicting creep rupture are based on internally pressurized tubes, the NuScale steam generator tubes are externally pressurized. As a result, the calculated probability of a thermally induced SGTF is judged to be overestimated because creep progresses more vigorously under tension than under compression. The nominal temperature and stress conditions that the tubes are exposed to are derived from a representative MELCOR severe accident simulation. Uncertainty is accounted for by imposing a distribution about the nominal values for temperature, pressure, and the Larson-Miller parameter. The probability of such a failure is incorporated into the Level 2 PRA as described in Section 19.1.4.2. In the Level 2 PRA, if a core damage event causes a thermally-induced SGTF with concurrent failure of the secondary-side isolation valves on the damaged SG, a containment bypass accident has occurred and a large release is assumed. A thermally induced SGTF does not pose a unique severe accident phenomena risk that would threaten the CNV, and is not analyzed deterministically.

19.2.3.3.7 Other Severe Accident Mitigation Features

The NuScale design includes additional features that are relevant to mitigation of severe accidents. In addition to the capabilities summarized in the prior sections, the design includes unique features that are not explicitly credited in the PRA:

- Partial immersion of the CNV in the reactor pool provides radionuclide scrubbing in the event of CNV lower head failure.
- For severe accidents with CNV bypass or containment isolation failure, the release would potentially be further reduced by the Seismic Category I Reactor Building, which includes a spray system.

19.2.3.3.8 Equipment Survivability

Consistent with SECY-90-016, SECY-93-087 and SECY-94-302, equipment required to mitigate severe accidents is evaluated to perform its intended severe accident functions. As stated in the references, the evaluation is intended to demonstrate that there is reasonable assurance that equipment needed for severe accident mitigation and post-accident monitoring and sampling will survive in the severe accident environment. Severe accident environmental conditions may produce extremes in pressure, temperature, radiation, and humidity.

Following a severe accident in which core damage has occurred, the two functions that must be maintained are containment integrity and post-accident monitoring.

Equipment is qualified to 100-percent humidity. In terms of post-accident dose, the NuScale design has used a methodology for environmentally qualifying equipment in the containment based, in part, on severe accidents as modeled in the NuScale PRA. This approach provides confidence that the monitoring equipment remains functional following a severe accident. Post-accident temperature and pressure conditions are discussed with regard to containment integrity and post-accident monitoring capabilities as follows.

Containment Integrity