SUNI Review Complete
Template=ADM-013
E-RIDS=ADM-03

ADD: Michael Eudy, Kevin Hsueh, Sean Meighan, Mark Blumberg, Bridget Curran, Mary Neely
Comment (10)
Publication Date: 4/21/2022
Citation: 87 FR 23891

PUBLIC SUBMISSION

As of: 6/23/22 6:41 AM
Received: June 21, 2022
Status: Pending_Post
Tracking No. l4o-9gjn-ggad
Comments Due: June 21, 2022
Submission Type: Web

Docket: NRC-2021-0179
Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants

Comment On: NRC-2021-0179-0001
Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

Document: NRC-2021-0179-DRAFT-0012
Comment on FR Doc # 2022-08519

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General Comment

NuScale Power, LLC Submittal of Comments on Draft Regulatory Guide DG-1389, Proposed Revision 1 to Regulatory Guide 1.183: “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Docket ID NRC-2021-0179

Attachments

LO-120251_DG-1389 RG_Alt Reg Source Terms Eval Design Basis Accidents_signed

June 21, 2022 Docket No. NRC-2021-0179

Office of Administration
Mail Stop: TWFN-7-A60M
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
ATTN: Program Management,
Announcements and Editing Staff

SUBJECT: NuScale Power, LLC Submittal of Comments on Draft Regulatory Guide DG-1389, Proposed Revision 1 to Regulatory Guide 1.183: "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Docket ID NRC-2021-0179

REFERENCES:

1. "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," 87 Fed. Reg. 23891, April 21, 2022.

2. Draft Regulatory Guide DG-1389, Proposed Revision 1 to Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," April 2022 (ML21204A065).

In a Federal Register Notice dated April 21, 2022 (Reference 1), the U.S. Nuclear Regulatory Commission (NRC) issued for public comment the document DG-1389 (Reference 2), requesting that comments be submitted no later than June 21, 2022.

The attachment to this letter provides comments of NuScale Power, LLC (NuScale) on DG-1389.

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

If you have any questions, please contact Mark Shaver at 541-360-0630 or at mshaver@nuscalepower.com.

Sincerely,

Carrie Fosaaen
Director, Regulatory Affairs
NuScale Power, LLC

Attachment: "NuScale Power Comments, U.S. Nuclear Regulatory Commission Draft Regulatory Guide DG-1389, Proposed Revision 1 to Regulatory Guide 1.183: 'Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors'"

NuScale Power Comments U.S. Nuclear Regulatory Commission DG-1389 Proposed Draft Rev 1 of Regulatory Guide 1.183 “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"			
Comment #	Section	Comments/Basis	Recommendation
1.	Section A, Purpose, et al	The document purpose states that “this guide establishes an AST based in part on SAND-2011-0128, “Accident Source Terms for Light Water Nuclear Power Plants Using High Burnup of [*sic*] MOX Fuel….” Section 1.1 states “as used in this guide, the AST is an accident source term that is derived principally from SAND-2011-0128 and differs from the TID-14844 and NUREG-1465 source terms used in the original and revised design and licensing of operating reactor facilities.” Section B, Background, states “Revision 0 of RG 1.183 will continue to be available for use by licensees and applicants as a method acceptable to the NRC staff for demonstrating compliance with the regulations.” Taken together these comments imply that Revision 1 applies to high-burnup fuel (only) while Revision 0 continues to apply to low-burnup applications. The RG should be clarified to discuss the scope of SAND-2011-0128 and its use within RG 1.183 (covering both LBU and HBU applications) and the relationship to and continued availability of NUREG-1465 as implemented by Revision 0.	The Purpose, Reason for Revision, and Background should be clarified to discuss the scope of SAND-2011-0128 and its use within RG 1.183 (covering both LBU and HBU applications) and the relationship to and continued availability of NUREG-1465 as implemented by Revision 0. The statement that Revision 0 remains available should clarify the rationale for such, and that continued availability should be elevated beyond the “background” portion of the RG. Section A, Applicability, should explicitly define applicability to both LBU and HBU applications, and note continued availability of Revision 0.

Comment #	Section	Comments/Basis	Recommendation
2.	Section A, Purpose and Applicability	The document purpose states that "this guide establishes an AST based in part on SAND-2011-0128, "Accident Source Terms for Light Water Nuclear Power Plants Using High Burnup of [*sic*] MOX Fuel…." Section 3.2 states that the release fractions are not endorsed for MOX fuel. The applicability of this RG for MOX fuel should be addressed up front in the Purpose and Applicability discussions of Section A.	Address RG applicability for MOX fuel in in the Purpose and Applicability of Section A.
3.	Section A, Applicable Regulations, Page 2, Bullet 2	The first sentence of the second bullet identifies the five active subparts of 10 CFR 52: early site permits (ESP), standard design certifications (DC), combined licenses (COL), standard design approvals (SDA), and manufacturing licenses. The second sentence discussing the requirement to evaluate offsite radiological consequences only discusses two of the five subparts: DC (52.47) and COL (52.79). It is not clear why only these two subparts are addressed and the other three are omitted. ESP (52.17), SDA (52.137), and ML (52.157) all contain equivalent requirements.	Include references to 10 CFR 52.17, 10 CFR 52.137, and 52.157 in the second sentence of the second bullet.

Comment #	Section	Comments/Basis	Recommendation
4.	Section B, Reason for Revision, Page 5	The reason provided for revision of the guide is to address new issues since the original guide was issued. The issues are then listed. Although the list is helpful to understand what changed, in the case of technical changes to the previous guidance, no information is provided as to the need for the specific changes—e.g., it is not clear if the changes were made to address non-conservatisms in the old guidance or to eliminate over-conservatisms in the old guidance. Although the Revision 0 of RG 1.183 will remain available for use, a user needs to be apprised of the purpose of the changes in order to determine which version is appropriate for their use. As an example, the new guidance makes small changes to some release fractions and large changes to others. Some of the changes are increases and some are decreases. The intent of the overall net impact of these release fraction changes is not clear. It would also be helpful to know whether the cumulative impact of all the changes in the new guidance would be expected to calculate an increase or decrease in dose relative to the old guidance.	For each of the listed technical changes, the reason for revision should be clearly stated. The reasoning should indicate whether the new guidance is more or less conservative than the prior guidance. If the new guidance is simply more accurate than the old guidance and both are conservative, then that should be stated.

Comment #	Section	Comments/Basis	Recommendation
5.	Section B, Page 5, Footnotes 1 & 2	Footnote 1 provides a definition of a maximum hypothetical accident (MHA). In the definition, the MHA is also referred to parenthetically as a maximum credible accident (MCA). The MHA has historically been referred to as the MCA in some contexts (e.g. TID-14844), however the context of the statement might lead users to believe that the MHA LOCA is itself "credible." Further, the purpose of the MHA in relation to other regulatory requirements should be clarified.	Revise footnote 1 to clarify. Suggested revision, in marked-up form, is as follows: The maximum hypothetical accident (MHA) ~~(also referred to as the maximum credible accident)~~ is that accident whose consequences, as measured by the radiation exposure of the surrounding public, would not be exceeded by any accident whose occurrence during the lifetime of the facility would appear to be credible (note: historical references such as TID-14844 may refer to the MHA as the "maximum credible accident," although this terminology does not indicate the MHA is itself credible). As used in this guide, the term "LOCA" refers to any accident that causes a loss of core cooling. The MHA LOCA refers to a loss of core cooling resulting in substantial meltdown of the core with subsequent release into containment of appreciable quantities of fission products. Although the design basis of a facility's ECCS is required to prevent a LOCA from resulting in substantial meltdown of the core, the MHA evaluation is conducted to assess the performance of containment, fission product control systems, and site characteristics in providing defense-in-depth against a more severe accident than allowable within the plant's design basis. These evaluations assume containment integrity with offsite hazards evaluated based on design basis containment leakage.

Comment #	Section	Comments/Basis	Recommendation
6	Section B, Background, Page 6, Paragraph 3	The third paragraph contrasts the whole body and thyroid dose criteria of 10 CFR 100.11 with the total effective dose equivalent criteria in 10 CFR 50.34, 10 CFR 52, and 10 CFR 50.67. In addition to those regulations, 10 CFR 100.21 provides a connection from Part 100 to 10 CFR 50.34 for newer plants. It may be beneficial to identify 10 CFR 100.21 in the list of applicable regulations.	Add a reference to 10 CFR 100.21 where applicable.
7.	Section C, Subsection 1.1	Section 1.1 states “ASTs may be used for advanced LWRs under 10 CFR Part 50 and 10 CFR Part 52 and for operating reactors under 10 CFR 50.34 and 10 CFR 50.67.” “Advanced” in this context seems to refer to new reactors; “advanced” implies a more limited applicability of ASTs.	Replace “advanced” with “new.”
8.	Section C, Subsection 1.3.1, Pages 11&12	Subsection 1.3.1 provides a list of regulatory requirements. The text indicates that the list is not all inclusive. However, the list neglects to include significant references, such as 10 CFR 50.34(a)(1) and 10 CFR 100.21.	Add 10 CFR 50.34(a)(1) and 10 CFR 100.21 to the list of regulatory requirements.
9.	Section C, Subsection 1.3.1, Page 12	Item g of the list in Subsection 1.3.1 is related to 10 CFR 52. However, the item only includes three of the five active subparts of 10 CFR 52 (ESP, DC, COL). The other two active subparts of 10 CFR 52 (SDA and ML) should be added to Item g.	Add SDA and ML to Item g.

Comment #	Section	Comments/Basis	Recommendation
10.	Section C, Subsection 3.1, Page 17, Paragraph 1	Subsection 3.1 states that the core inventory should be determined using an appropriate isotope generation and depletion computer code. The previous revision of this RG provided examples of computer codes that met this guidance. It is not clear why the names of the computer codes have been removed. Does the removal of the examples imply that the computer codes in the previous revision of the RG are no longer adequate? Or is the intent of eliminating the examples to provide flexibility to use other computer codes?	Clarify the guidance regarding what is an appropriate computer code for depletion.
11.	Section C, Subsection 3.2, Page 19	Subsection 3.2 provides tables of release fractions. These tables are easy to use and relatively similar to the previous revision. However, the text indicates that the actual release should be based on the tables plus any T_{FGR} prompted by the accident conditions. Correlations are provided to determine T_{FGR} as a function of burnup and increase in radial average fuel enthalpy. The guidance indicates that non-LOCA DBAs such as fuel handling accidents should consider T_{FGR}. Transient analyses are typically not performed for fuel handling accidents and increase in enthalpy is not applicable. There is no guidance provided for determining T_{FGR} for fuel handling accidents. The guidance for T_{FGR} appears to be incomplete.	Clarify the applicability of T_{FGR} to the various design basis events which consider dose consequences. Provide guidance for determining T_{FGR} for fuel handling accidents.

Comment #	Section	Comments/Basis	Recommendation
12.	Section C, Subsection 3.3, Page 22	Subsection 3.3 provides a table with the onset time and duration of release phases. Table 5 indicates that the duration of the gap release phase is 0.22 hours for PWRs and 0.16 hours for BWRs. The paragraph of text below Table 5 concludes with a statement that the duration of the gap release phase is 0.5 hours. This appears to be an inconsistency that causes confusion about what gap release duration should be used.	Clarify the relationship between the durations in Table 5 and the durations in the text. Clearly identify the applicable duration.
13.	Section C, Subsection 4.2, Page 24	Subsection 4.2 introduces the concept of “transit dose”. The guidance indicates that the licensing basis for some licensees may include transit dose and that those licensees should include the transit dose when evaluating the 5 rem TEDE control room dose limit associated with 10 CFR 50.67. The concept of transit dose is not found in the supporting regulations . The regulatory basis for this concept is not provided. It is also not clear why it would be applicable to some licensees and not to others. In addition, the only guidance provided for evaluation of this transit dose is “on a case-by-case basis.” This RG is intended to also provide useful information for new reactor applicants and this paragraph addressing transit dose ends with a statement that new reactor licensees may use “this guidance” in demonstrating CR habitability. Therefore, it is important to clarify whether evaluation of transit dose is necessary, with appropriate regulatory basis, and if so to provide prescriptive guidance on including transit dose. Further, clarify whether the statement that new reactor licensees “may use this guidance” refers to the preceding discussion of transit dose or to the remaining discussion of Subsection 4.2.	The regulatory basis, if any, for the concept of transit dose should be provided. If there is none, then the concept should be clearly confined to those existing licensees for which transit dose is part of their licensing basis. If there is a basis to apply the guidance to new reactor applicants, then the guidance should provide prescriptive instructions for evaluating transit dose. In addition, the end of the paragraph should clarify the meaning of “this guidance” with respect to new reactor licensees.

Comment #	Section	Comments/Basis	Recommendation
14.	Section C, Subsection 4.2.3, Footnote 15, Page 25	Footnote 15 states that the nuclides used for modeling dose from airborne radioactivity inside the control room may not be conservative for determining the dose from radioactivity outside the control room. No guidance is provided for how it is determined whether or not the nuclide set is conservative. Similarly, no guidance is provided for determining an alternative set of nuclides that would be conservative. If it is not acceptable to use the set of nuclides from inside the control room, then guidance should be provided for what is acceptable.	Footnote 15 should be removed or else revised to provide guidance as to what is an acceptable set of nuclides for determining dose outside the control room.
15.	Section C, Subsection 5.3, Page 31, Paragraphs 1&2	Subsection 5.3 states that RG 1.145 and another reference document methodologies used in the past for determining atmospheric dispersion (X/Q) values. The previous revision of this RG provided examples of computer codes that met this guidance. It is not clear why the names of the computer codes have been removed. In addition, it is not clear what is meant by methodologies “used in the past”. It seems to imply that the methodologies are out-of-date and possibly should not be used in future. If this is not the intent, the guidance should be clarified to indicate that these methodologies “used in the past” are still acceptable for current and future use.	Clarify guidance regarding what is an appropriate computer code for dispersion. Clarify whether previous guidance in RG 1.145 is still acceptable.

Comment #	Section	Comments/Basis	Recommendation
16.	Appendix B, Subsection B-2, Page B-2	Subsection B-2 provides very detailed guidance for determining the decontamination factor (DF) for the fuel handling accident. The very detailed guidance has a very narrow applicability of water depths (19 to 23 ft). No guidance is given for depths less than 19 ft or greater than 23 ft. The previous guidance gave a bounding DF which could be used for all depths greater than or equal to 23 ft. As a result, many licensees and new reactor applicants rely on design features to ensure that the depth is at least 23 ft. Typically, the depth is larger than 23 ft, as more water is thought to be better for shielding purposes, etc. Under the new guidance, licensees and new reactor applicants with a depth of greater than 23 ft would have to justify the selection of DF "on a case-by-case basis". This new guidance appears to encourage licensees and new reactor applicants to decrease the water depth to less than 23 ft in order to fall in the applicable range of the equations. This seems non-conservative. The old guidance, which provides a bounding DF for depths greater than 23 feet, is much more straightforward and easy to apply. In addition, the equation provided for depths between 19 and 23 feet is a complicated function of pin pressure that is not easily comparable to the old guidance. The equation has a discontinuity around 5000 psig. For pin pressures less than 5000 psig, the minimum DF calculated by the equation appears to be approximately 250. If the new equation always yields values greater than 250 (for pin pressures less than 5000 psig), then the guidance should indicate that a value of 250 may be used in lieu of calculating pin pressures and using the equation. This would also confirm that the prior guidance was conservative.	The guidance should provide a bounding DF that can be used as an alternative to calculating DF based on pin pressure. The guidance should also be expanded to include depths greater than 23 ft, as this is the most common depth designed for by licensees and new reactor applicants.

Comment #	Section	Comments/Basis	Recommendation
17.	Appendices E & F	Appendices E and F are provided for the PWR SGTR and PWR MSLB events, respectively. In the previous revision of the guidance, these two Appendices were exactly reversed, with Appendix E being for the PWR MSLB event and Appendix F being for the PWR SGTR event. The Appendices for the event-specific guidance for all other events is maintained consistent between the two revisions. It is not clear the benefit of switching the order of Appendices E and F. In addition, it may cause confusion to licensees and applicants who are accustomed to the Appendices being aligned to specific events.	Reverse the order of Appendices E and F so that consistency with the order of the previous revision is maintained.
18.	Appendix F, Subsection F-6.6, Page F-3	In the prior guidance, the MSLB transport model was based on a figure included with the MSLB appendix. The figure was similar to the figure provided as Figure E-1 in the SGTR appendix of the new guidance. The new guidance is not clear on whether the new Figure E-1 applies to the MSLB transport model or not. If it does apply, the cross-reference should be included. If it does not apply, no explanation is provided for why it no longer applies.	Clarify whether the new Figure E-1 applies to the MSLB transport model or not and provide either the applicable cross-reference or the explanation for why it no longer applies.