	Part 1: Comments on Pre-decisional DG ML 22276A149, "Technology-Inclusive Risk-Informed, and Performance-Based Methodology for Seismic Design of Commercial Nuclear Plants"			
	Section	**Comment/Basis**	**Recommendation**	**SwRI Team Response**
1	General	The basic idea is good–some safety-related SSCs could be designed to a lesser seismic design criterion, which would make the plant less expensive and, if applied properly, just as safe.	None	Thanks
2	Introduction B. Proposed Options D. Implementation	In the advanced reactor public meeting that took place on 10-12-2022, NRC explained that the guide's Options 2 and 3 can also be used under part 50 as long as: • A singular SSE applies and its value is not below per Part 50 Appendix S • Safety criteria similar to those of Part 53 Framework A are clearly defined if the SDC is below a Category 5.	Consider addressing how to apply Options 2 and 3 to Part 50.	This is now outside the scope of the two Draft RGs. NRC staff to decide whether a separate RG could be developed to realize these options.
3	B. Proposed Options	The DG provides an example for Option 2, but not for Option 3. Option 3 may be the most desirable, but it is also the one that will be hardest to reach consensus on without further guidance.	Consider providing a similar example for Option 3.	Develop one or more example applications of Option 3 (with specifications for Framework A and Framework B) as Appendix B to Draft RG 1410.
4	B. Proposed Options	The problem with Option 2 is the requirement to perform a "generic" SPRA. The DG is light on content, so one needs to refer to the RIL since the DG points to it. Here are some thoughts: a. The plants are not designed to a uniform hazard response spectrum, but to a CSDRS that the vendor	See Comment Basis	The information and analyses necessary to address these comments are important, but too detailed and extensive to include in the Draft RG

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		selects to hopefully "bound" the sites where they might build (or at least yield a design that can be qualified for those sites even if the curve is exceeded to some extent at certain frequencies). b. The RIL suggests that one approach could be to select a bounding hazard. There are two parts: b.i. Bound the spectral shape. This means to take the actual spectral shape for each SDC at all (or a subset) for the existing plant sites in the US (there bounds the set at all frequencies. This would not be a UHRS, so would not actually provide an "accurate" SPRA for any site (a point that the RIL acknowledges but does not make a judgement on). b.ii. Bound the PSHA. This appears to mean to take the hazard curve for the same sites and draw a shape that bounds the AEF at each pga level. This further compounds the issue that it would not apply to any site, and doesn't correspond to the spectral shape being used. b.iii. By definition, this should be adequately conservative to assure that the categorization would work at any site that was included in the set of sites used for the bounding process. b.iv. This "double bounding" approach will almost assuredly result in "over-categorization" of SSCs for most designs. The issue is that you are matching a PSHA (frequency) curve with a spectrum that is likely quite irrelevant and very conservative (e.g., the bounding PSHA yields a high pga that is mostly		1410. We propose to expand on all these points as we revise the RIL and turn it into a NUREG/CR that provides many of the technical bases for the two draft RGs.

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		associated with sites with low spectral acceleration, but the bounding spectrum has high spectral accelerations associated with sites with low pga). b.v. It may also miss some things, since site-specific issues such as liquefaction, subsidence, SSI, etc. could be missed. This could be an issue later, since it is also required to perform the final SPRA and confirm the categorization. At this point it is too late – the plant has been designed (so it is not too late to change something, but that defeats the purpose of standardized design). c. The RIL suggests a second approach, which would be to perform what is effectively a series of PRAs by using the UHRS, PSHA, and site-specific conditions for multiple sites. The implication is you then categorize the SSCs based on the most restrictive (which will likely come from different sites). This would likely give a "better" answer, but also be quite a bit more expensive to implement. It may result in less "over-categorization" and be less likely to be challenged by a site-specific PRA. d. Either approach requires selection of an adequate range of sites to envelope where a plant might be built. Otherwise, it could result in a "failure" to confirm the categorization at a particular site.		
5	B. Proposed Options	The key problem with Option 3 is that the DG is very light on content and the RIL doesn't discuss it at all. This is highly problematic. There needs to be some actual guidance on what would be acceptable,	See Comment Basis	Some minor clarification has been added in this draft. More

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		otherwise it may take so long to reach agreement that it may not be timely enough to be useful. a. Option 3 does not require a SPRA, either for the categorization or for the confirmation. It allows "other risk-informed approaches" but does not provide any indication of what these might be. b. Categorization could be based in large measure on the consequences of the failure of the SSC. For some designs, this approach would be very desirable. b.i. Again, there needs to be some indication regarding the selection of the scenarios. b.ii. There also needs to be some guidance on what would be acceptable for determining that the resulting consequences would not exceed the F-C curve if the design criteria were relaxed. There would need to be some consequence margin based on some judgment of the potential accident frequency but taken too far it approaches Option 2.		enhanced discussion will be included in Appendix B and revised RIL, which we will publish as a NUREG.
6	B. Proposed Options	Both Option 2 and Option 3 require that the design decisions be "confirmed". For option 2, this is done with SPRA. For Option 3, either by SPRA or other risk-informed approaches. The DG does not actually discuss what this means, but the implication is that this is the site-specific analysis associated with a COL or construction permit application. The RIL doesn't mention this at all. Again, the lack of any guidance brings up concerns because once the design is done you would not want to have to change an SSC from, say SDC-4, to SDC-5. There needs to be some finality to the design requirement that would	See Comment Basis	Discussions in the planned NUREG/CR will be expanded to clarify that the use of a F-C curve is not mandatory in Option 3. However, both Framework A and Framework B ultimately require

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		not result in a categorization change based on the specific site where a plant would be built. a. Would it be adequate to show that the overall seismic risk is low, or would you also have to show that the F-C for each seismic scenario that includes an SDC-4 SSC is under the curve? b. Would you have to demonstrate that there are no SDC-4 SSCs that are "risk-significant" regardless of the overall or individual scenario risk?		a SPRA type analysis.
7	Introduction B. Proposed Options	Based on the advanced reactor public meeting that took place on 10-12-2022, NRC also discussed the possibility to use the options under Framework B as long as one adopts safety criteria from Framework A instead of Framework B's principal design criteria. These clarifications allow more flexibility to the industry and should be covered in the DGs otherwise the industry may assume that if they use Framework B or Part 50, they are not allowed to use the methodology from the DG.	Consider addressing how to apply Option 2 and 3 to Framework B.	The revised draft RGs now address both Framework A and Framework B. However, the draft RGs do not include any 10 CFR Part 50 discussions. A reference to AERI will be added to the Draft RGs in the May revision.
8	Introduction	It may be useful to address somewhere, not necessarily in these guides, what NRC will require to see if exemptions are considered for either Part 53 or Part 50/Part 100 seismic requirements. NRC has proposed in the past that some advanced reactors may need to take exemptions from Parts 50/100. Designers in the industry are considering such exemptions for their design, beyond the alternatives offered in the DG.	Consider the best way to provide this insight to the industry.	NRC staff will decide whether a separate RG could be developed to realize these options under 10 CFR Part 50 or Part 52.

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		Therefore, it may be useful to address this pathway as well as to what NRC needs to see to approve potential exemptions. Such exemptions may be related to the SSE/DBGM, and site specific seismic/geotechnical investigations/analyses which are not required for some nuclear facilities in the U.S (see comment under "B Proposed Option" below for context).		
9	Related Guidance	RG 1.232. 1.143, 1.166 and NUREG-0800 are mentioned in the corresponding section of the seismic isolator DG but not in this DG.	Add as needed.	These references are now included in both revised draft RGs.
10	B. Discussion	ASCE 43-19 will be endorsed based on this DG while 43-05 is endorsed based on RG 1.208. It is not clear if NRC plans to update RG 1.208 to change the ASCE revision or if the industry can continue to rely on the 43-05 version when operating under Part 50.	Consider different revisions of ASCE 43 relied upon by the NRC and clarify if the latest version only should be followed.	NRC plans to revise RG 1.208 because the current version of RG 1.208 only refers to SDC-5. The two draft RGs reference ASCE 43-19 to cover SDC-2 to SDC-5. If NRC staff choose to provide separate guidance on 10 CFR Part 50, this topic will be included there.

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11	B. Proposed Options	It appears that there should be an option 4 that follows NRC's process for review and approval of facilities such as medical isotope facilities but applied to advanced reactors that have a similar risk level due to their low inventory and dose consequence. Shine medical and Northwest medical clearly detail in their PSAR in Chapter 2 that they rely solely on the seismic methodology of IBC/ASCE 7. Despite these facilities processing radioisotopes under Part 50 licensing, their low-risk thresholds allow the NRC to approve their application despite it utilizing industrial non-nuclear codes. IBC/ASCE 7 are also the codes used for DOE for its nuclear facilities. It is unclear why the same cannot be done for micro-reactors that present a very low risk and can prove it through a source term analysis. It is understood that IBC/ASCE 7 could be used as part of Option 3, however it is not clear if sufficiently low source term results based on an extreme accident would be sufficient.	Consider an option for the use of IBC/ASCE 7 based on medical isotope and DOE nuclear facility precedents.	The scope of the two draft RGs is based on ASCE 43-19. The IBC/ASCE 7 approach to characterizing the hazard and evaluating SSCs differs from that in ASCE 43-19 and current NRC guidance. An evaluation of ASCE 7 for use with commercial nuclear power plants has not been performed by the NRC staff or industry for its applicability to lower-risk facilities. We propose a technical meeting on this topic to address this comment.
12	B. Proposed Options C.2.4	The following statement is somewhat vague:	NRC should elaborate on this and provide	Some minor clarifications have been added in this

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		"risk insights; and a combination of approximate, bounding, and conservative analyses and quantitative risk information" "other risk-informed approaches" Several reactor designers will be interested in how to comply with this.	examples. The Appendix may be a candidate for this.	draft. More enhanced discussion, including some examples, will be included in Appendix B and the revised RIL/NUREG.
13	Proposed Options	Echoing previous comments, Option 1's methodology is laid out in RG 1.208, while Option 2 is backed up by NEI 18-04's documentation. Something similar is needed to support Option 3, so it is clear what is acceptable. This would help avoid back and forth with the NRC which could be due to the industry's interpretation of what is reasonable for Option 3.	Preferably, the requested clarifications for Option 3 should be part of this DG but could be incorporated in an external document.	Some minor clarifications have been added in this draft. More enhanced discussion, including some examples, will be included in Appendix B and the revised RIL/NUREG
14	C.2 and C.3	Option 3 is still tied to NEI-18-04 because of the emphasis on IDP. The connection to IDP is confusing.	Option 3 should be revised to better clarify that using Option 3 is not the same as applying NEI 18-04 (and its corresponding IDP).	Some minor clarifications are planned for the next version. More enhanced discussion, including some examples, will be also included in Appendix B and

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				the revised RIL/NUREG
15	C. Staff Pre-decisional Appendix A	The document references RIL 2021-04 which states: "The approach in this report can also accommodate codes other than ASCE 43, such as ASCE 7 (ASCE/SEI, 2010), for the design of low-risk facilities. (Microreactors are special systems of relatively low risk, and the regulatory framework for them is evolving (see, e.g., BNL, 2020)." "The process uses ASCE 43, for consistency with the rest of this report; however, compliance with risk criteria may be shown using other design codes, such as ASCE 7." "It is envisioned that some of the very small advanced reactors and microreactors (with negligible offsite consequences) can be seismically designed in accordance with ASCE 7. The LMP would impose an additional and undue burden in the reactor design process".	Considering these assertions from the NRC and previous comments, NRC should generate guidance that is specific to ASCE 7 and its use, taking into account how it is already implemented for medical isotope facilities.	The scope of the two draft RGs is based on ASCE 43-19. The IBC/ASCE 7 approach to characterizing hazard and evaluating structures and systems is entirely different than ASCE 43-19 and current NRC guidance. It would require an in-depth evaluation to assess its applicability to lower-risk facilities. Based on NRC direction, ASCE 7 may be referenced in future revisions to the two draft RGs.
16	C.3.1	Some SDCs below 5 are discussed as examples; however, alternative SDCs were also discussed in the advanced reactor public meeting that took place on 10-12-2022 as a possibility. Technically both SDC	It would be helpful to explain that SDC 2 and 1 can be attained based on	SDC-1 is outside the scope of ASCE 43. Therefore, SDC-1

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		2 and 1 are possibilities depending on the dose consequence of the reactor.	the overall risk of the facility.	is not considered in this guide. Although, one can imagine other SDCs, ASCE 43 provides specific guidance for SDCs -2, -3, -4, and -5, assuring the target performance in a risk-graded approach is achieved.
17	C.3.3	The section states "Seismic loads are prescribed as OBE or one-half SSE." However, OBEs greater than 1/3 SSE per 10 CFR 50 Appendix S require additional work and it is not clear if it is also the case for this DG. Appendix S states: "A value greater than one-third of the Safe Shutdown Earthquake Ground Motion design response spectra. Analysis and design must be performed to demonstrate that the requirements associated with this Operating Basis Earthquake Ground Motion in Paragraph (a)(2)(i)(B)(I) are satisfied. The design must take into account soil-structure interaction effects and the duration of vibratory ground motion."	Consider adding some clarification considering the 1/3 SSE stipulated in Appendix S.	Discussion will be expanded in the revised proposed NUREG/CR. However, we conclude that an OBE specific design is impractical when multiple design basis motions are used. For proposed Part 53, we do not include an OBE design option.

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18	C. Staff Pre-decisional	There is a lot of discussion of peer reviews related to this section and it is not clear how relevant/required that is when codes other than ASCE 43 and ASCE 4 are relied upon, or when a non-PRA method is used per Option 3.	Consider what clarification might be needed in light of the comment that supports the introduction to other codes and standards.	This topic will be considered for latest versions as per NRC direction. However, both draft RGs describe initiatives that are being proposed for the first time in the nuclear industry. Many situations will likely arise that will require judgement and review by experts, for example non-linear analysis.
19	Appendix A B. Proposed Options	ASCE 349 and N690 are mentioned in Appendix A. There may be cases where these codes are not used as part of Option 3. In those cases, their commercial/industrial counterparts ASCE 318 and ASCE 360 would be used. This would be the case when the IBC/ASCE 7 is used.	The guidance should be updated to clarify that the use of alternative codes is not only applicable to the substitution of ASCE 43 and 4 as called out in Figure 1 and extends to other nuclear codes.	The NRC will consider the applicability of ASCE 7 after we have an opportunity to assess its applicability. In the proposed approach it is crucial that an SSC achieves a design target that allows it to

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				perform its needed safety function and meet safety criteria. The current version of the two draft RGs do not include application of IBC/ASCE 7.

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	Section	**Comment/Basis**	**Recommendation**	**NRC Response**
1	General	All comments from the above table that are also applicable to this DG should also be considered and are not repeated here.	Make consistent updates as needed	Understood
2	Related Guidance	It may be worthwhile to call out the NUREG series on seismic isolation NUREG/CR-7253 to 7255 in this section.	Add as needed.	We agree and we made this revision.
3	Background	It may be good to also mention the ITER fusion reactor under construction in addition to the Horowitz reactor in France as it also uses seismic isolation based on NUREG/CR-7255. This may become relevant when NRC generates regulation for fusion reactors.	Add as needed.	We agree and we made this revision.
4	Background	This section states “ASCE/SEI 43-19 is the only consensus standard to provide criteria for the seismic design criteria for applications of SI systems in nuclear facilities.” However, as noted in the previous table some nuclear facilities rely on ASCE 7. ASCE 7 2021 has seismic isolation provisions in Ch 17 would be suitable for nuclear structures where the dose consequence justifies the use of that code under Option 3.	Consider modifying background to account for the broader reference on seismic isolation in other standards.	The scope of the Draft RG 1307 is based on ASCE 43-19. The IBC/ASCE 7 approach to characterizing hazard and evaluating structures and systems is entirely different than ASCE 43-19 and current NRC guidance. It would require an in-depth evaluation to assess its applicability to lower-risk facilities. Based on NRC direction, ASCE 7 may be referenced in future revisions to the two draft RGs.

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	Section	**Comment/Basis**	**Recommendation**	**NRC Response**
5	Figure 2	The figure mentions “Advanced LWR” under Option 1, however not all advanced reactors are LWRs.	The language should be made technology inclusive.	We agree and we modified the text accordingly.
6	Figure 2	Given that at least ASCE 7 also provides criteria for seismic isolators and testing, the requirement to follow the testing requirement from 43-19 and 4-16 in cases where ASCE 7 would be acceptable is not fully justified and may be an undue burden.	Propose making note conditional and less restrictive or remove note as it is implied that the testing should be in accordance with the codes used.	In the current version of the draft RG, only ASCE 43-19 is considered. An applicant can always use a different code if adequate justification demonstrates that the intent of the technical positions is achieved. We will add this language back into the draft RGs in the May revision.
7	Figure 2	The graded SPRA and RG 1.233 being the primary requirements for Option 3 is a departure from the DG documented in the previous table. However later discussions related to Option 3 state: “Option 3 based on a broad spectrum of approaches that include deterministic inputs, risk insights, and a combination of \approximate, bounding or conservative analyses, and quantitative risk information.”	Consider adding the flexibility of “and Other Risk-Informed Approaches” for Option 3. This would be consistent with C.3.4.	The reference to RG 1.233 was removed and some clarifying language was added to the draft RG text.
8	Figure 2	Based on earlier comments in this table and the previous one, it is not reasonable to require all seismically isolated advance reactors to use ASCE 43-19 and 4-16.	Consider adding the flexibility of “and other codes and standards with proper justification”.	In the current version of the draft RG, only ASCE 43-19 and 4-17 are considered. An applicant can always use a different code if

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			This would be consistent with C.3.3.	adequate justification demonstrates that the intent of the technical positions is achieved.
9	Figure 3	It is also important to note that if other codes are allowed, earthquake recurrence and performance targets need to be commensurate with those codes and the selected SDC. This impacts the performance targets and criteria discussed for DBE at 10^{-5} and BDBE at 10^{-6}, probability of unacceptable performance is to be less than 1% and 10% under DBE DBGM and BDBE DBGM well as the "167% of the DBGM" noted in the figure. Note also that requirements to also consider performance of the isolators during BDBEs could lead to over-designing the isolators.	Consider similar recommended relaxation as above.	We will consider this in possible future revisions to the draft RG. However, to move in this direction, NRC will need additional technical information. We propose a technical meeting to discuss how to align other codes and standards and to define performance targets.
10	General	NUREG/CR-7253 offers: "The ground motion response spectrum…are calculated for design of nuclear power plants… by multiplying the ordinates of a uniform hazard response spectrum at the specified hazard exceedance frequency by a design factor that is greater than or equal to 1.0. **The factor can be** seismically isolated nuclear **set equal to 1.0** for design of a power plant if the **earthquake risk is dominated by horizontal ground shaking and a stop is provided**..." This is understood to mean that the DF utilized in RG 1.208 to convert the UHRS into the GMRS is effectively 1 when using isolators where the	Consider including this fact.	We are taking this into consideration. NUREG/CR-7253 ASCE 43-19 will be reviewed to determine whether and how we can modify the draft RG as requested, for the planned May revision.

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		horizontal seismic force control and adequate CS is provided. This is an important fact that is not discussed in the DG.		
11	Figure 2	Option 2 later discusses 1.5 x DBE but that is not mentioned in the figure. PGA is discussed for this DG and the previous, however when other codes are used per Option 3, PGA may not be the key seismic input. When ASCE 7 is used, PGA informs the foundation design, however the design of SSCs and the DRS is based on bounding spectral acceleration rather than PGA.	Consider clarifying this similar to what was done for Option 3 in the figure. Consider if the focus on PGA might be misleading.	We are considering the clarification, however, the discussion of the 1.5 factor in Option 2 is partly in reference to when contact loads should be considered in a design. As stated earlier, the discussion in this guide is related to ASCE 43-19. Both the regulations and ASCE 43 characterize design basis motion by a DRS. The use of PGA to describe the DRS is a historical practice. PGA can be related to another descriptor of a DRS.
12	C.3.4	This statement appears to provide some flexibility but the RG 1.233 occurring after "other risk-informed approaches" still makes it restrictive: "The applicant should demonstrate that the final design satisfies Part 53 safety criteria which could be accomplished through a graded SPRA (or other risk-informed approaches) in accordance with the guidance provided in RG 1.233."	Consider making even restrictive. less	We agree and we have modified the text in the current version of the draft RG.

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13	C.4.5 C.4.6	The PRISM design referenced in the DG called for a qualification program including the following to determine horizontal static and dynamic stiffness, vertical stiffness, damping and margin to failure/failure modes: • The testing of high damping bearings • The qualification of expansion joints for the secondary heat transfer system piping • Large building tests with prototype isolators • Scale model tests of reactor structure with isolators on a shake table • The development of analytical models • Bearing material optimization and qualification • The development of seismic isolation guidelines • • Seismic margin assessment	Consider clarifying whether any of this still applies in case it is not captured by latest to be endorsed versions of ASCE 43/4 (when they are used).	This topic will be evaluated and expanded on the revised RIL/NUREG.
14	General	Overall Option 3 which should provide the most flexibility is still tied to the used of LMP guidance which is not the case for Option 3 discussed in the previous table.	Consider previous recommendations on this to give more flexibility under Option 3 and optionally decouple it from LMP, NEI 18-	We agree with decoupling it from the LMP, NEI 18-04/IDP, and RG 1.233. We modified the text in the current version of the draft RG to reflect this. However, ASCE

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			04/IDP, RG 1.233 and ASCE 43/4.	43-19 is still the primary code we rely on for the reasons discussed in the guide.