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A FRAMEWORK FOR SAFETY DECISION MAKING GIVEN NEW SEISMIC HAZARD INFORMATION

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ABSTRACT

Knowledge and understanding of natural hazards is continually evolving. As a result, there is a need to re-examine the safety of nuclear facilities in the light of new knowledge and reassess the adequacy of original design bases. Department of Energy (DOE) nuclear facilities must comply with DOE Order 420.1C Facility Safety, which requires that facilities review their natural phenomena hazards (NPH) assessments at least every ten years. Re-evaluation programs have also been undertaken for nuclear power plants (NPPs) and significant post-Fukushima re-evaluation activities are underway in the United States (US). Once completed, all US NPPs will have updated seismic hazard information and a significant number will have completed seismic probabilistic risk analysis (SPRAs).

Information from DOE facilities, new information on NPP capacity, and the maturing of probabilistic seismic hazard analysis (PSHA) and SPRA techniques provides a unique opportunity to develop a new, risk-informed framework that can be used to effectively address technical and regulatory questions that arise from new natural-hazard information. A risk-informed approach for re-evaluating seismic hazard and risk information for a broader range of facilities would benefit the NRC, DOE, the nuclear industry, and the public. It could also serve as a template for addressing a large suite of other natural hazards.

BACKGROUND

Knowledge and understanding of natural hazards is continually evolving. As a result, there is a need to re-examine the safety of nuclear facilities in the light of new knowledge and reassess the adequacy of original design bases. While several important re-evaluation programs have been undertaken for both DOE- and NRC-regulated facilities, the maturing of PSHA and SPRA techniques provides an opportunity to develop a new risk-informed framework that can be used to effectively address technical and regulatory questions that arise from new natural hazard information. A framework for periodically assuring seismic safety could also serve as a template for addressing a large suite of other natural hazards.

DOE nuclear facilities must comply with DOE Order 420.1C Facility Safety, which requires that all such facilities review their NPH assessments no less than every ten years and evaluate the need for an update. The Order points to criteria in Standard DOE-STD-1020-2012. This Standard also references other documents such as Standard ANSI/ANS-2.29-2008 and NUREG-2117 (NRC, 2012a). These documents provide supporting criteria and approaches for evaluating the need to update an existing PSHA. All of the documents are consistent at a high level regarding the general conceptual criteria to be considered. However, none of the documents provides step-by-step detailed guidance on the required or recommended approach to evaluating whether or not an existing PSHA should be updated. Further, all of the conceptual approaches and criteria given in these documents deal only with hazard-related changes.

Given that the DOE Order is aimed at ensuring the *safety* of nuclear facilities—which is a function not only of seismic hazard but also the seismic capacity of the facility—the inclusion of risk information in the evaluation process is in line with the spirit and objectives of the Order. Recently, Idaho National Laboratory (INL) undertook efforts to define a more risk-informed and rigorous approach to comply with the DOE Order (INL, 2015a). An Independent Review Panel, composed of the authors of this paper, was formed to, "validate, review, and refine the risk-informed methodology" that was proposed. The review panel developed a white paper (INL, 2015b)[1] that advanced the INL proposal and provides generalized guidance for seismic design category (SDC) 3 to 5 facilities (ANS/ANSI-2.26-2004;R2010).

The need for a new framework for NPPs is particularly acute given the significant post-Fukushima re-evaluation activities underway in the US. Once completed, all US NPPs will have updated seismic hazard information and a significant number will have completed SPRAs. These plants will have demonstrated, and in some cases been modified to ensure, appropriate safety margin accounting for updated hazard information. There will be an unprecedented understanding of demonstrated plant capability, as well as risk insights concerning the major contributors to plant-specific seismic risk. The generalized approach for SDC-5 facilities is aligned with current NRC activities and can be used for reevaluation activities in the future. Although focused on the reevaluation requirements of the DOE Order, the framework can also be applied whenever significant new seismic information is available.

This paper proposes and discusses the elements of a risk-informed framework for facilities that takes advantage of both mature probabilistic analysis techniques and the significant amount of experience using them. Elements of the framework are summarized here, but are more fully discussed in (INL, 2015b). This white paper also includes guidance on determining if existing risk studies are appropriate for use and discusses considerations for choosing the method of risk analysis to be applied.

APPLICABILITY

The framework presented is consistent with the existing US approaches. It assumes that a viable point of comparison for new hazard information exists. The safety and risk assessment elements of the framework can be applied at facilities that do not have a PSHA. However, a Senior Seismic Hazard Analysis Committee (SSHAC) Level 3[2] PSHA study following NUREG/CR-6372 (NRC, 1997) and NUREG-2117 should be conducted.

The process is intended to address requirements related to periodic reevaluation or evaluation of new seismic information that may challenge current design or licensing bases of nuclear facilities. This framework is not intended to replace the seismic hazard analysis requirements in Standard ASME/ANS RA-Sb-2013, although the Standard can fit within the processes proposed. Because INL is located in the western US, the framework was originally developed for sites that do not have an endorsed regional model; however, some guidance on the comparison of new information against an endorsed regional model is provided in (INL, 2015b).

The framework proposed is for existing facilities. New construction or a major modification of a facility requires a SSHAC Level 3 (SL3) study be conducted and up-to-date design and risk analysis methods be used, thus obviating the need for the framework in this document. Likewise, a decision to conduct a SL3 study can be made at the outset or at any time. This would also obviate the need to further evaluate the

[1] The white paper that forms the basis of the discussions herein is in draft form at the time of this writing and is subject to change. The reader should refer to the final version of (INL, 2015b), once published.

[2] As discussed in NUREG 2117, SSHAC Level 3 and 4 studies are considered to provide the same level of regulatory assurance. Throughout this document, reference is made to SSHAC Level 3, which has become strongly preferred in the US. However, a SSHAC Level 4 study could be used where a Level 3 is noted.

existing hazard study, although the subsequent risk studies and comparisons may be needed, depending on the purpose of the review.

EVALUATION CRITERIA

Seven criteria (summarized below) are used in the framework. The first four criteria are from DOE-STD-1020-2012 and are supported and amplified in ANSI/ANS-2.29-2008 and NUREG-2117. These address the inputs to the hazard analysis and the hazard results themselves. Criteria #5 through #7 are proposed to provide additional risk-focused considerations. These seven criteria are applied slightly differently for SDC-3, SDC-4, and SDC-5 facilities due to the differing radiological risk of the categories. Because a confident basis is needed for applying the criteria, a site-specific SSHAC Level 1 (SL1) or 2 (SL2) study is conducted as a starting point. The study should include (an assessment of associated uncertainties and in-process peer review).

Criterion #1 requires that all relevant new and updated data, models, and methods be identified, compiled and evaluated to determine if relevant new information is available beyond that used for the original PSHA.

Criterion #2 requires that appropriate changes to the PSHA model be determined and implemented based on the activities associated with Criterion #1. Important inputs to the hazard model include those required for the seismic source characterization, ground motion characterization, and near-surface site response analysis.

Criterion #3 evaluates whether or not the technical bases for the PSHA model have changed, even if the calculated hazard has not changed. The technical bases include the technical arguments and justifications for the hazard inputs and the associated treatment of uncertainties.

Criterion #4 compares the original hazard results with estimated results from an updated model, accounting for the precision with which mean seismic hazard is typically calculated.

Criterion #5 compares the mean hazard at the hazard exceedance probability specified for the SDC category (Table 1) to the facility's design basis ground motion (DBGM).

Table 1: Target performance goals, probability ratios, and hazard exceedance probabilities recommended for design of nuclear facilities (summarized from Table 2-2 of ASCE/SEI-43-05).

Earthquake Design Parameters			
	Seismic Design Category (SDC)		
	3	**4**	**5**
Target Performance Goal (P_F)	1 x 10^{-4}	4 x 10^{-5}	1 x 10^{-5}
Probability Ratio (R_P)	4	10	10
Hazard Exceedance Probability (H_D)	4 x 10^{-4}	4 x 10^{-4}	1 x 10^{-4}
Note: H_D=R_P x P_F			

Criterion #6 is similar to #5, but the basis for comparison is a Ground Motion Response Spectrum (GMRS), as defined for SDC-5 NPPs in NRC Regulatory Guide (RG) 1.208 (NRC, 2007). The GMRS is developed based on a uniform hazard spectrum (UHS) at an annual frequency of exceedance (AFE) of risk significance coupled with "Design Factors" (DF) that account for the slope of the hazard curve at the

AFE. The GMRS was used by the NRC as a risk-informed screening measure for the post-Fukushima evaluation of all US NPPs (NRC, 2012c).

The GMRS in RG 1.208 defines a level of ground motion for design that will confidently ensure that risk levels of NPPs are within acceptable limits. The definition of GMRS for SDC-3 and SDC-4 categories must also achieve the appropriate target performance goals. Unfortunately, at the time of this writing, modification factors for SDC-3 and SDC-4 facilities have not been published. However, the upcoming revision of ASCE 43 is expected to provide "Scale Factors" (SF) similar to the DF in ASCE 43-05. Until these SF are published, there are three options: (1) do not calculate the GMRS and follow the "no" path from Criterion #6 to Criterion #7, (2) develop and justify a site-specific GMRS following the concepts in ASCE 43-05, or (3) use GMRS processes for SDC-5 facilities. Facilities outside the US could make use of this approach for determining GMRS, provided that their facilities have similar seismic capacities (and can meet similar performance criteria). Alternately, country-specific factors could be developed.

Criterion #7 is a comparison of the facility-specific risk information against risk-informed criteria. Risk-informed criteria are both quantitative (e.g., performance goals in ASCE/SEI-43-05) and qualitative (e.g., diversity of risk-significant equipment and accident sequences). The criteria discussed in (INL, 2015b) are specific to US regulatory frameworks. However, other countries can make use of this approach by defining the specific basis for risk-informed decision-making consistent with their own risk-informed regulatory frameworks.

USE OF NEW OR EXISTING SPRA OR MARGINS INFORMATION

Incorporating the estimated hazard information into an appropriate SPRA can provide risk estimates and insights necessary for risk-informed decisions (Standard ASME/ANS RA-Sb-2013 and NRC, 2009). This information includes estimates of mean risk values and uncertainty distribution, dominant contributors, potential accident sequences, and potential consequences. Use of an appropriate existing or new seismic margin assessment (SMA) study can also be an appropriate alternative to SPRA in some cases, as discussed in detail in (INL, 2015b).

The use of an existing SPRA or SMA study, in conjunction with the new hazard, is a quick way to determine the impact of new hazard information on the assessed risk of the facility. However, certain considerations and criteria have to be satisfied in order to use an existing analysis and model. As discussed in detail in (INL, 2015b), these considerations and criteria are related to the continued validity of the existing model in terms of how well it reflects current conditions of a facility and the model's ability to realistically estimate facility response to the latest ground motion characteristics. Results from a SPRA or a well-executed SMA study can be used to identify the most safety-significant enhancements. However, "interim actions" may also be prudent when new estimates of ground motion are significantly higher than the DBGM and the SPRA or SMA is yet to be completed. Assessment of the need for interim actions is incorporated into the framework and is discussed in greater detail in (INL, 2015b).

Throughout the process, the assessment of the available information against the criteria requires engineering judgment by knowledgeable experts. The SL1 or SL2 Technical Integration team needs to be a small team of people in order to ensure that various technical disciplines are sufficiently represented and that all relevant information is identified and incorporated. An in-process peer review of the SL1 or SL2 study is also required. The risk assessment elements of the process should similarly be conducted by a team of qualified individuals with the appropriate knowledge and experience. An in-process peer review is recommended for the risk studies, consistent with other guidance (NRC, 2013d; EPRI, 2013a; and ASME/ANS, 2013).

As discussed in (INL, 2015b), a project-level peer review is also recommended to ensure that the overall risk-informed decision process whitepaper achieves its objectives. It would be appropriate and efficient for the project-level peer review team members to act as reviewers for the individual elements (e.g., a hazards expert could act as a reviewer for the initial SSHAC Level 1 or 2 study) to ensure continuity and a broad understanding.

PROPOSED APPROACH TO RISK-INFORMED DECISION MAKING

Facilities Categorized as SDC-3

The decision steps for exercising Criteria #1 through #7 are shown in Figures 1 to 3. The approach for an SDC-3 facility (Figure 1) starts by using the information from the SL1 or SL2 study to assess whether or not new or updated information has become available since the existing PSHA was conducted (Criterion #1). If no new data, models or methods are available, then it is assumed that there would also be no change to the hazard analysis or the hazard results. Therefore, the only action taken involves documentation of the approach used, the outcome, and the basis for the findings. If new information is available, then an evaluation must be made as to whether or not the hazard inputs would change from the previous PSHA (Criterion #2). If the inputs do not change, that outcome would be documented as above. If changes in the hazard inputs are anticipated, then changes in the calculated hazard value itself would next be calculated and considered (Criterion #4). If no significant changes in hazard levels are anticipated, an assessment would next be made of whether or not the technical bases for the hazard and its various inputs have changed (Criterion #3). If so, it is prudent to continue the decision process, as emphasized in DOE-STD-1020-2012. If neither the calculated hazard values nor the technical basis for the assessment have changed, no further action is required beyond full documentation.

The next decision step (Criterion #5) is the comparison of the ground motions associated with SDC-3 AFE (H_D=4x10^{-4}) with the existing design basis ground motions for the facility of interest. If the DBGM exceeds the SDC-3 H_D ground motions, then one moves to Criterion #6. If the DBGM is not greater than the SDC-3 motions, then it is recommended that additional risk and/or safety evaluations be performed. There should also be an evaluation of the need for interim safety improvements until more detailed risk evaluations are conducted.

Criterion #6 is a comparison of the existing DBGM with the calculated GMRS. If the DBGM for the facility exceeds the GMRS, then one documents the decision process and no further actions are necessary. If the GMRS exceeds the existing DBGM, it is recommended that additional risk and/or safety evaluations be performed. For those cases where the DBGM has been found to be exceeded by either the applicable SDC-3 ground motions or the GMRS, additional risk analyses are conducted. The first risk-informed decision point is Criterion #7 in which the mean risk results and the risk insights are assessed against risk-informed criteria.

In the case where all risk objectives are met, the study and outcomes are documented and no further actions are required. Where the risk objectives are not met, there are two possible cases. First, the calculated risk values may exceed the performance goal by a large margin or risk insights may highlight an area of significant concern. If so, an "immediate safety issue" is identified and immediate actions are determined. This is followed by a SL3 study, which could be used as input to design modification and any associated engineering changes. If exceedance of the risk objectives is small, there is no need to identify an immediate safety issue, but a SL3 study is conducted. Additional actions can include modifications or procedural changes, as necessary. They can also include updates to the design or regulatory bases, as necessary.

Facilities Categorized as SDC-4

The process for SDC-4 facilities is shown in Figure 2. The steps, decision criteria, and recommended actions for the first six criteria are the same as for SDC-3. However, if all of the first four criteria are met and the SDC-4 ground motion or GMRS exceeds the DBGM, then a SSHAC Level 3 study is conducted. The results of the PSHA are used as input to a SPRA or SMA study. Risk insights and risk estimates are reviewed against risk objectives in a similar manner to the SDC-3 methodology. If the risk objectives are met, then the study and outcomes are documented and no further action is taken. If the risk objectives are not met, then the subsequent actions are the same as those given for SDC-3 facilities.

Facilities Categorized as SDC-5

The process for SDC-5 facilities is shown in Figure 3. The decision steps incorporated into this recommended approach are consistent with those implemented as part of the post-Fukushima requirements from the NRC in their 50.54(f) letter (NRC, 2012b). They are also consistent with a risk-informed decision process that is endorsed by DOE, the Defense Nuclear Facilities Safety Board, and the NRC. If the application of the first four criteria results in “yes” findings, then it is recommended that a SL3 PSHA be conducted to replace the existing PSHA.

The mean hazard at the design AFE (H_D=1x10^{-4}) from the new PSHA and the GMRS will be compared with the existing DBGM. If the GMRS exceeds the DBGM, then seismic risk studies will be conducted. The additional risk evaluations using the updated hazard will be used to evaluate the need for interim safety improvements to be conducted in the near term. The results of the risk studies will be compared to the target performance goals and risk objectives. The comparison of whether or not risk objectives have been met and the subsequent actions are the same as those recommended for SDC-3 and SDC-4 facilities, and include flagging the assessed risk as an immediate safety issue or proceeding with changes in the DBGM, design bases, and/or retrofits that will lower the seismic risk from the facility to acceptable levels.

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DISCLAIMERS

The views in this paper are those of the authors. The views expressed in this paper do not necessarily reflect the view of the INL, DOE, NRC or any other entity.

REFERENCES

ANSI/ANS (American National Standards Institute/American Nuclear Society). 2004 (Reaffirmed 2010). Categorization of Nuclear Facility Structures, Systems, and Components for Seismic Design. ANSI/ANS-2.26-2004; R2010, Le Grange Park, Illinois.

ANSI/ANS (American National Standards Institute/American Nuclear Society). 2008. Probabilistic Seismic Hazard Analysis. ANSI/ANS-2.29-2008, Le Grange Park, Illinois.

ASCE/SEI 43-05, Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities, 2005, Structural Engineering Institute of the American Society of Civil Engineers.

ASME/ANS (American Society of Mechanical Engineers/American Nuclear Society). 2013 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications. Addendum B to Standard ASME/ANS RA-Sa-2013, 2013.

DOE (U.S. Department of Energy). 2012a. DOE Standard, Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities. DOE-STD-1020-2012, Washington, D.C.

DOE (U.S. Department of Energy). 2012b. Facility Safety. DOE Order 420.1C, Washington, D.C.

Electric Power Research Institute (EPRI). 2013. Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic. EPRI Report 3002000704. May 2013. Palo Alto, CA.

Idaho National Laboratory (INL, 2015a), "Project Execution Plan PLN-31287: INL Seismic Risk Assessment Project," Idaho National Laboratory Report, Idaho Falls, 2015.

Idaho National Laboratory (INL, 2015b), "Proposed Methodology for Evaluating the Need for an Update of an Existing PSHA to Comply with DOE Order 420.1C", INL Seismic Risk-Informed Methodology Independent Review Panel," Idaho National Laboratory Report, Idaho Falls, 2015.

Kennedy, R. P. (2011). Performance-goal based (risk informed) approach for establishing the SSE site specific response spectrum for future nuclear power plants. Nuclear Engineering and Design, 241(3), 648-656.

NRC (U.S. Nuclear Regulatory Commission). 1997. Recommendations for probabilistic seismic hazard analysis: guidance on uncertainty and the use of experts. NUREG/CR-6372, two volumes, US Nuclear Regulatory Commission, Washington, D.C.

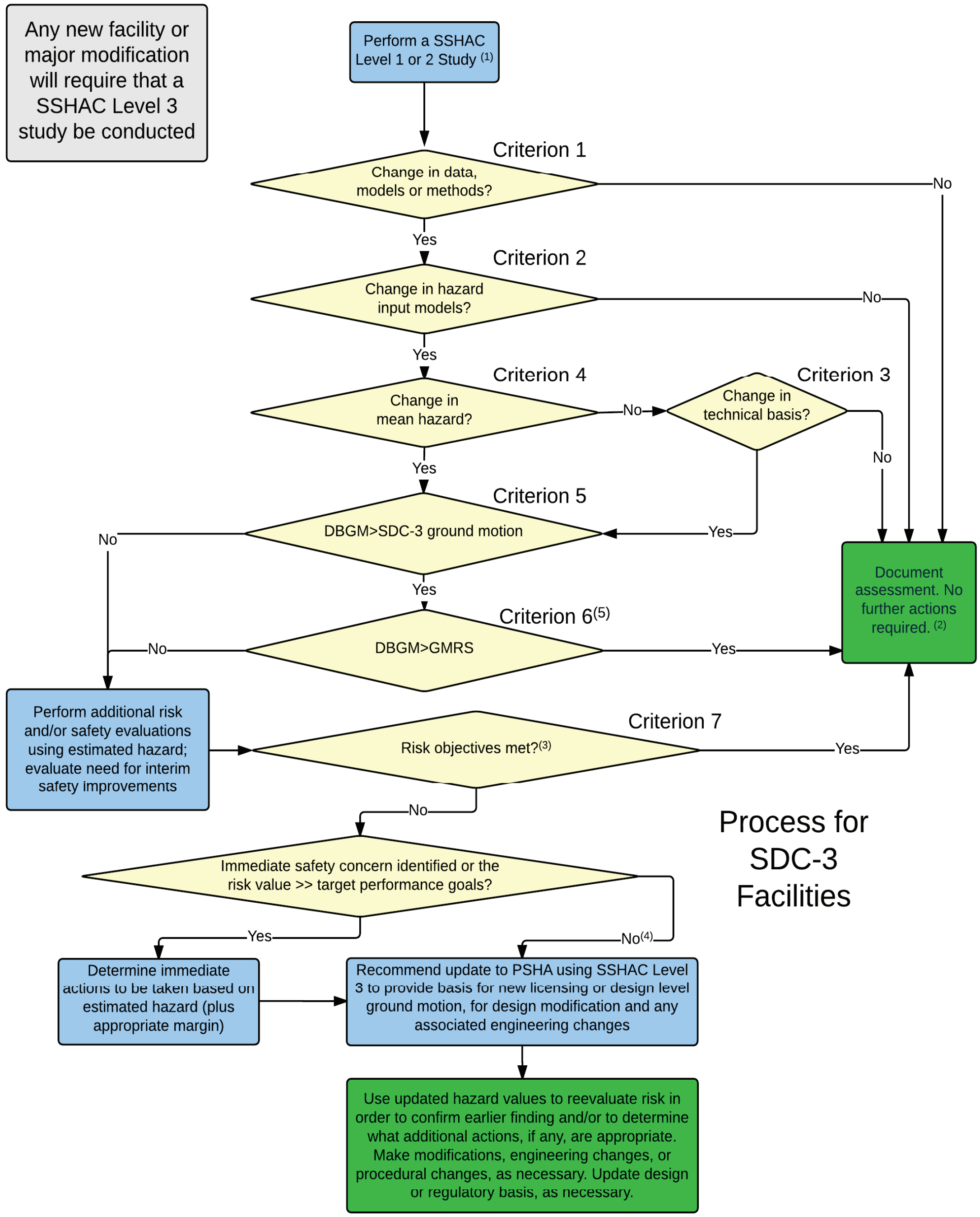
NRC (U.S. Nuclear Regulatory Commission). 2007. A performance-based approach to define the site-specific earthquake ground motion. Regulatory Guide 1.208, Washington, D.C.

NRC (U.S. Nuclear Regulatory Commission). 2009. An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities. Regulatory Guide 1.208, Washington, D.C.

NRC (U.S. Nuclear Regulatory Commission). 2012a. Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies. NUREG-2117, Rev. 1, Washington, D.C.

NRC (U.S. Nuclear Regulatory Commission). 2012b. Central and Eastern United States Seismic Source Characterization for Nuclear Facilities. NUREG-2115, Washington, D.C.

NRC (U.S. Nuclear Regulatory Commission). 2012c. "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) regarding Recommendation 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident." Dated March 12, 2012, Washington, D.C.



Notes:
1. The SSHAC level 1 or 2 study must include a documented in-process peer review. The resulting estimated hazard should be compared with the existing studies used to develop design or regulatory bases for the facility.
2. The estimated hazard and any risk-informed findings apply to this facility only and cannot be used for other facilities
3. See the report for discussion of the appropriate risk-related information to be considered
4. A "no" determination here indicates that the risk objectives were not shown to be met, but that no immediate safety concern was identified and the risk did not exceed the target by a large amount.
5. Currently design factors needed for developing the GMRS have only been published for SDC-5 Facilities (ASCE 43-05). The next revision of ASCE 43 is expected to have the necessary "scale factors" for SCD-3 and SDC-4 facilities.

Figure 1. Chart showing the proposed methodology for evaluating the need for an update of a PSHA for a SDC-3 facility.

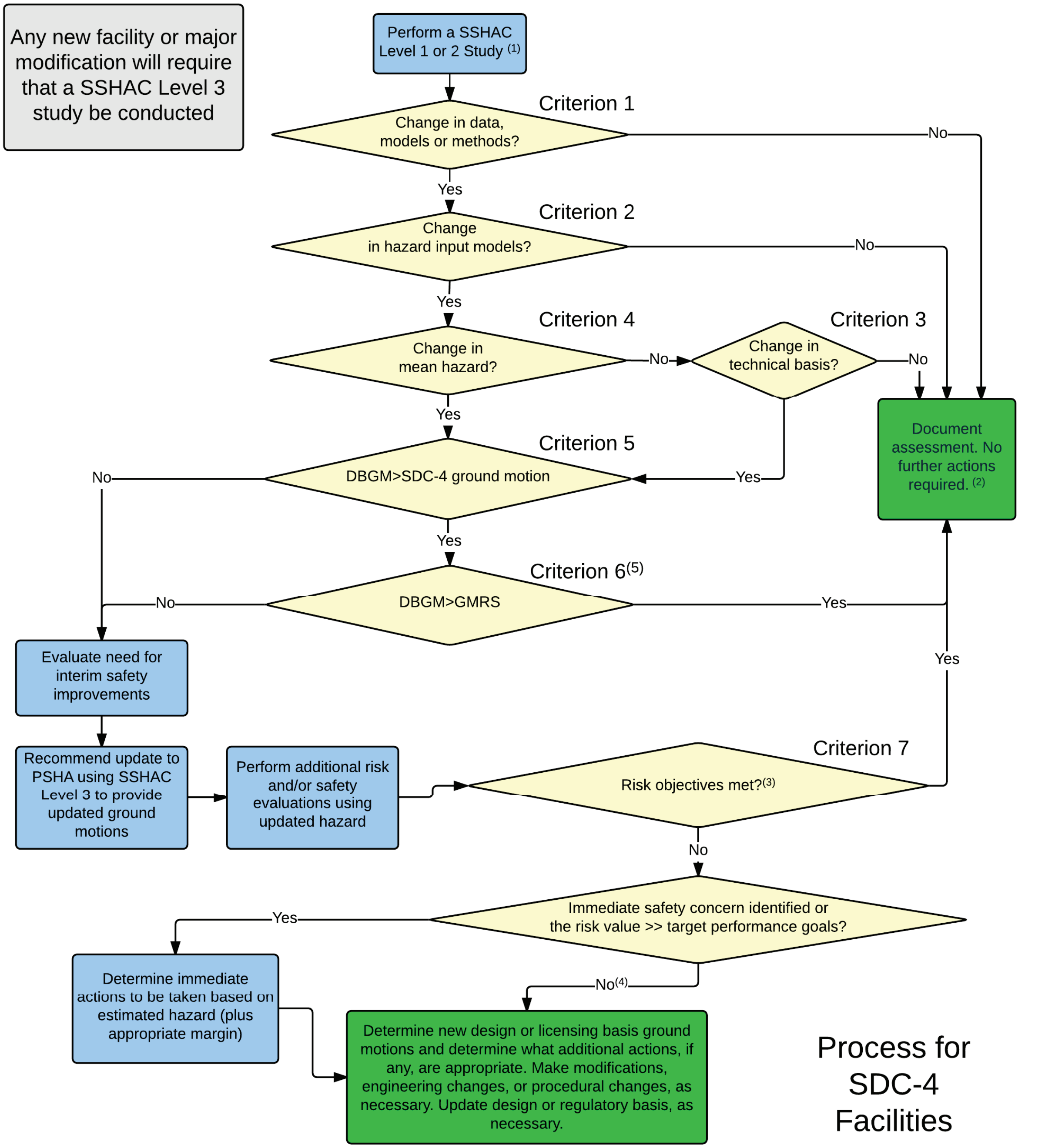


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Figure 2. Chart showing the proposed methodology for evaluating the need for an update of a PSHA for a SDC-4 facility.

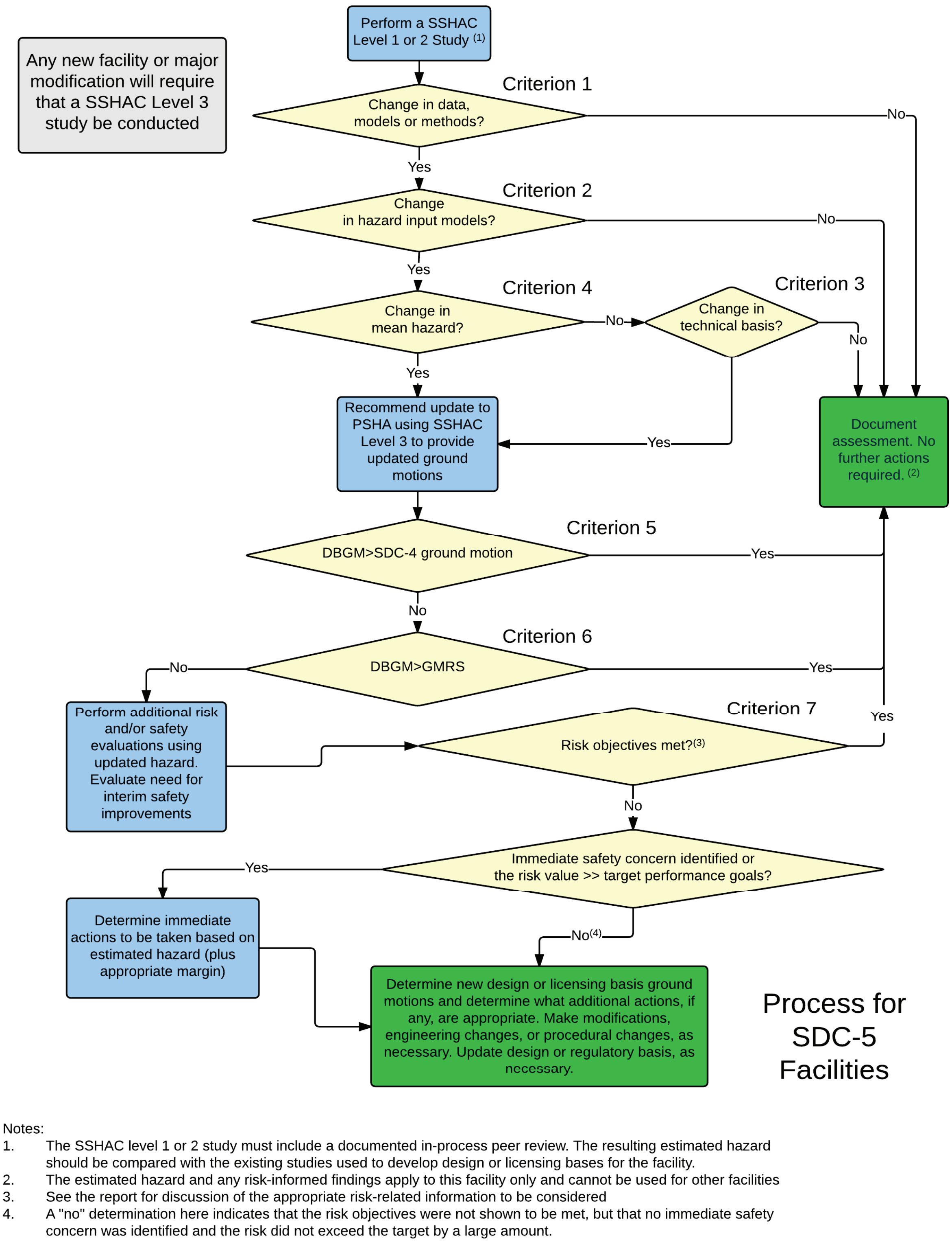

Figure 3. Chart showing the proposed methodology for evaluating the need for an update of a PSHA for a SDC-5 facility.