ASME/ANS RA-S-1.3-2017

Standard for Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications

TRIAL USE AND PILOT APPLICATION

Publication of this Standard for trial use has been approved by The American Society of Mechanical Engineers and the American Nuclear Society. Distribution of this Standard for trial use and comment shall not continue beyond 24 months from the date of publication, unless this period is extended by action of the Joint Committee on Nuclear Risk Management. It is expected that following this 24-month period, this draft Standard, revised as necessary, will be submitted to the American National Standards Institute (ANSI) for approval as an American National Standard. A public review in accordance with established ANSI procedures is required at the end of the trial-use period and before a Standard for trial use may be submitted to ANSI for approval as an American National Standard. This trial-use Standard is not an American National Standard.

Comments and suggestions for revision should be submitted to:

Secretary, Joint Committee on Nuclear Risk Management
The American Society of Mechanical Engineers
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New York, NY 10016-5990

The American Society of Mechanical Engineers

ANS

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(This Foreword is not part of "Standard for Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications," ASME/ANS RA-S-1.3-2017)

FOREWORD

The American Nuclear Society (ANS) Standards Board and the American Society of Mechanical Engineers (ASME) Board on Nuclear Codes and Standards (BNCS) mutually agreed in 2004 to form a Nuclear Risk Management Coordinating Committee (NRMCC). This committee was chartered to coordinate and harmonize standards activities related to probabilistic risk assessment (PRA) between the two standards developing organizations (SDOs). A key activity resulting from the NRMCC was direction to the ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM) to develop PRA standards structured around the three Levels of PRA (i.e., Level 1, Level 2, Level 3) to be jointly issued by the two societies.

This Standard sets forth requirements for determining consequences (i.e., Level 3, also referred to as L3 in this Standard) as part of PRAs and related analysis methodologies that can be used to support risk-informed decisions for commercial nuclear power plants. This Standard also prescribes a process for applying these requirements for certain other applications involving release of radioactive materials into the atmosphere (e.g., non-light water reactor (LWR) nuclear power plants, research reactors, fuel cycle facilities, and non-reactor nuclear Department of Energy (DOE) facilities). In these cases, supplemental requirements may be needed to ensure technical adequacy.

This Standard was developed based on the body of knowledge and experience accumulated through the development and application of the ASME/ANS RA-Sb-2013, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," and Level 2 PRA Standard ASME/ANS RA-S-1.2-2014, "Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs)," which has been approved for trial use and pilot application. This Standard, however, is not dependent upon these other PRA standards, although it is noted that the development of the final risk estimation for reactors will be based on combining the results of the Level 1 and Level 2 (Level 1/2) PRA portions (e.g., release frequencies, release characterizations) and the results of the consequence analysis.

Consequences covered within the scope of this Standard include radiation dose and induced health effects, and economic impacts, taking into account atmospheric dispersion, demography, dosimetry, pathways to man, and plant/site characteristics. The radioactive source terms and their frequencies often are passed on from Level 1/2 analyses.

The scope of a PRA covered by this Standard is primarily targeted for use to determine the impact of an accident at a nuclear power plant. However, the technology discussed here can be used to determine the impact of a release of radioactive material from any facility. A Level 3 analysis can use the results of a Level 1 analysis followed by a Level 2 analysis or the results of a combined Level 1/2 analysis (e.g., gas-cooled or other advanced reactors).

This Standard describes requirements for calculating the consequences of radionuclide releases into the environment and how to present the results of such calculations. It is assumed that a computerized consequence model will be used. Therefore, emphasis has been placed on the information that is typically required as input and available output. As with any computer code, there are pitfalls associated with its use, and there are uncertainties inherent in the quality and representativeness of the input data and the fidelity of the modeling. This Standard attempts to caution against improper use of consequence analysis tools.

This Standard contains a brief description of each major requirement to perform a consequence analysis, and explains why it is necessary, what information results, and how it is to be used. The technical requirements for the various technical elements of a consequence analysis include (1) transport and dispersion in the atmosphere; (2) deposition processes; (3) processes that lead to the accumulation of radiation doses; (4) protective measures, such as evacuation, that can reduce radiation doses; (5) the effects of radiation doses on the human body; and (6) economic impacts. A section is also included describing how the combined risk results of a Level 1, 2, and 3 PRA can be presented. This process is referred to as “risk estimation.”

It is acknowledged that some topics are subject to argument and continuing development, since consequence modeling is not a precise science and contains significant inherent uncertainties. Where an understanding of the current state-of-the-art is deemed necessary for a sensible interpretation of the results, a discussion of this topic is included. Other areas that are described in some depth are those in which the user's choice of input data can significantly affect the output. Examples include evacuation and sheltering, and dry deposition velocity.

Appendix A, Computer Codes, has been included in this Standard to provide some history and to illustrate typical input parameters and output reports of the calculation results from an acceptable computer code.

This Standard might reference documents and other standards that will have been superseded or withdrawn at the time the Standard is applied. A statement has been included in the reference section that provides guidance on the use of references.

The format for this Standard was developed in 2005 when no “standard” format was available. Therefore, it is not consistent with some other published PRA standards regarding chapter numbers. Following trial use, the format of the section numbering will be re-evaluated.

This Standard is issued for trial use and pilot application. Feedback is requested regarding the Standard in all areas including the following:

- Were the format changes that vary slightly from other contemporary PRA standards helpful? This includes descriptors added for each supporting requirement (SR).
- Were the technical SRs and action verbs clear?
- Notes have been included for a number of SRs. Do these notes result in lack of clarity regarding what is required and what is provided as added information? Are these notes helpful?
- Is the information provided in Appendix A useful?
- The bases for Capability Categories (i.e., Table 1-1) in this Standard differ from the other PRA standards in that two attributes are used (i.e., site specificity and model realism) rather than three attributes (i.e., scope and level of detail, plant specificity, and realism). It is thought that the scope and level of detail attribute is adequately addressed by the model realism attribute for Level 3 analyses, and that site specificity is more appropriate than plant specificity. Comments on this change are of interest.
- Capability Category III is expected to be deleted from this Standard (consistent with planned changes to the Level 1 and Level 2 PRA standards) following the trial use and pilot application period. Are there requirements in Capability Category III that should be considered for incorporation into Capability Category II rather than deletion?
- Some SRs contain multiple actions verbs (e.g., PA-B1, ME-A3). Did the inclusion of multiple action verbs in a single SR result in complications in meeting the requirements or assessing their completion as part of a Peer Review?
- Were uncertainty requirements easily understood and implemented?

- Were the minimum requirements for peer review teams reasonable (number of members, composition)?
- Was Section 5 on risk estimation used in your application, and if so were the requirements clear?
- The application process in Section 3 differs slightly from that of other PRA standards. Was the application process (e.g., flowchart in Figure 3-1) applicable (including references to Level 1 and Level 2 PRA scope)? If so did you have trouble applying the process?
- The ASME/ANS PRA standards have been developed in view of assessing the capability of a “base” PRA. It is recognized that nuclear facilities in the past have typically only developed Level 3 PRAs for specific applications, which may vary considerably, and were not maintained. Based on this historical usage of Level 3 PRA for specific applications, which may vary, this Standard has included some flexibility in the supporting requirements (e.g., no requirement for economic cost modeling or protective-action modeling for Capability Category I.) Are there areas where more or less specificity would be helpful in the supporting requirements in view of maintaining a “base” Level 3 PRA?
- A number of supporting requirements include examples. Are the included examples helpful, or do they create confusion as to what is required?

PREPARATION OF TECHNICAL INQUIRIES TO THE JOINT COMMITTEE ON NUCLEAR RISK MANAGEMENT

INTRODUCTION

NOTE FOR TRIAL USE: The text of this section describes the technical inquiry process for approved standards. However, during the trial use period, users are encouraged to provide feedback, ask questions, and interact with the Level 3 Working Group on either a formal or informal basis. Such feedback may be provided via the Secretary of the Joint Committee on Nuclear Risk Management, as noted below.

The ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM) will consider written requests for the interpretation and revision of risk management standards and the development of new requirements as dictated by technological development. JCNRM's activities in this latter regard are strictly limited to interpretations of the requirements or to the consideration of revisions to the requirements on the basis of new data or technology. As a matter of published policy, The American Society of Mechanical Engineers (ASME) does not "approve," "certify," "rate," or "endorse" any item, construction, proprietary device, or activity, and, accordingly, inquiries requiring such considerations will be returned. Moreover, ASME does not act as a consultant on specific engineering problems or on the general application or understanding of the standard's requirements. If, based on the inquiry information submitted, it is the opinion of the JCNRM that the inquirer should seek assistance, the inquiry will be returned with the recommendation that such assistance be obtained.

To be considered, inquiries will require sufficient information for JCNRM to fully understand the request.

INQUIRY FORMAT

Inquiries shall be limited strictly to interpretations of the requirements or to the consideration of revisions to the present requirements on the basis of new data or technology. Inquiries shall be submitted in the following format:

- *(a) Scope.* The inquiry shall involve a single requirement or closely related requirements. An inquiry letter concerning unrelated subjects will be returned;
- *(b) Background.* State the purpose of the inquiry, which would be either to obtain an interpretation of the standard's requirement or to propose consideration of a revision to the present requirements. Concisely provide the information needed for JCNRM's understanding of the inquiry (with sketches as necessary), being sure to include references to the applicable standard edition, addenda, part, appendix, paragraph, figure, or table;
- *(c) Inquiry Structure.* The inquiry shall be stated in a condensed and precise question format, omitting superfluous background information and, where appropriate, composed in such a way that "yes" or "no" (perhaps with provisos) would be an acceptable reply. This inquiry statement should be technically and editorially correct;
- *(d) Proposed Reply.* State what it is believed that the standard requires. If, in the inquirer's opinion, a revision to the standard is needed, recommended wording shall be provided;
- *(e) Typewritten/Handwritten.* The inquiry shall be submitted in typewritten form; however, legible, handwritten inquiries will be considered;
- *(f) Inquirer Information.* The inquiry shall include the name, telephone number, and mailing address of the inquirer;
- *(g) Submission.* The inquiry shall be submitted to the following address: Secretary, Joint Committee on Nuclear Risk Management, The American Society of Mechanical Engineers, Two Park Avenue, New York, NY 10016-5990.

USER RESPONSIBILITY

Users of this Standard are cautioned that they are responsible for all technical assumptions inherent in the use of PRA models, computer programs, and analysis performed to meet the requirements of this Standard.

CORRESPONDENCE

Suggestions for improvements to the Standard or inclusion of additional topics shall be sent to the following address: Secretary, Joint Committee on Nuclear Risk Management, The American Society of Mechanical Engineers, Two Park Avenue, New York, NY 10016-5990.

COMMITTEE ROSTERS

CONTRIBUTORS TO THE STANDARD FOR RADIOLOGICAL ACCIDENT OFFSITE CONSEQUENCE ANALYSIS (LEVEL 3 PRA) TO SUPPORT NUCLEAR INSTALLATION APPLICATIONS

(The following is a roster of the Joint Committee on Nuclear Risk Management at the time of the approval of this Standard.)

This Standard was processed and approved for release as a trial use and pilot application by the ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM). Committee approval of the Standard does not necessarily imply that all committee members voted for its approval. At the time it approved this Standard, the JCNRM had the following members:

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ASME/ANS RA-S-1.3 (formerly ANS/ASME-58.25 of the Standards Committee of the American Nuclear Society) was responsible for development of this Standard. The following is a list of members of the working group that provided input to the Standard:

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The ASME/ANS RA-S-1.3 Working Group wishes to provide special appreciation and recognition of the hard work, knowledge, and insights provided by Jocelyn Mitchell who passed away during the latter stages of the Standard development. Her guidance, support, contributions, and continued encouragement were keys to completing this Standard. She helped the group maintain appropriate balance of technical requirements through her continual scrutiny of superfluous additions, which were in her words, "gilding the lily."

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