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
Two Park Avenue
New York, NY 10016-5990

The American Society of Mechanical Engineers

ANS

ISBN 978-0-7918-7200-0

A 2 8 9 1 Q

9 780791 872000

Date of Issuance: July 13, 2017

ASME is the registered trademark of The American Society of Mechanical Engineers.

This code or standard was developed under procedures accredited as meeting the criteria for American National Standards. The standards committee that approved the code or standard was balanced to assure that individuals from competent and concerned interests have had an opportunity to participate. The proposed code or standard was made available for public review and comment that provides an opportunity for additional public input from industry, academia, regulatory agencies, and the public at large.

ASME does not "approve," "rate," or "endorse" any item, construction, proprietary device, or activity.

ASME does not take any position with respect to the validity of any patent rights asserted in connection with any items mentioned in this document and does not undertake to insure anyone utilizing a standard against liability for infringement of any applicable letters patent nor assumes any such liability. Users of a code or standard are expressly advised that determination of the validity of any such patent rights, and the risk of infringement of such rights, is entirely their own responsibility.

Participation by federal agency representative(s) or person(s) affiliated with industry is not to be interpreted as government or industry endorsement of this code or standard.

ASME accepts responsibility for only those interpretations of this document issued in accordance with the established ASME procedures and policies, which precludes the issuance of interpretations by individuals.

The American Society of Mechanical Engineers
Two Park Avenue, New York, NY 10016-5990

Published by

American Nuclear Society
555 North Kensington Avenue
La Grange Park, Illinois 60526 USA

Printed in the United States of America

CONTENTS

(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:

ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM)

R. J. Budnitz, *Co-chair*, Lawrence Berkeley National Laboratory
C. R. Grantom, *Co-chair*, C. R. Grantom P.E. & Assoc. LLC
D. W. Henneke, *Vice Co-chair,* General Electric Company
(Alternate: **Y. J. Li,** GE Hitachi Nuclear Energy)
P. F. Nelson, *Vice Co-chair*, National Autonomous University of Mexico

P. J. Amico, Jensen Hughes, Inc.
V. K. Anderson, Nuclear Energy Institute
R. A. Bari, Brookhaven National Laboratory
S. A. Bernsen, Individual
J. R. Chapman, Scientech, Inc.
M. Drouin, U.S. Nuclear Regulatory Commission
(Alternate: **D. E. Yeilding,** U.S. Nuclear Regulatory Commission)
K. R. Fine, FirstEnergy Nuclear Operating Company
K. N. Fleming, KNF Consulting Services LLC
H. A. Hackerott, Omaha Public Power District–Nuclear Energy Division
E. A. Hughes, Etranco, Inc.
G. W. Kindred, Tennessee Valley Authority
K. L. Kiper, Westinghouse Electric Company LLC
S. Kojima, Kojima Risk Institute, Inc.
G. A. Krueger, Exelon Corporation
(Alternate: **J. L. Stone,** Exelon Corporation)
S. H. Levinson, AREVA Inc.
S. R. Lewis, Electric Power Research Institute
(Alternate: **D. C. Hance,** Electric Power Research Institute)
A. Maioli, Westinghouse Electric Company LLC
J. O'Brien, U.S. Department of Energy
G. W. Parry, Jensen Hughes, Inc.
M. K. Ravindra, MKRavindra Consulting

M. B. Sattison, Idaho National Laboratory
R. E. Schneider, Westinghouse Electric Company LLC
B. D. Sloane, Jensen Hughes, Inc.
C. Spitzer, International Atomic Energy Agency
D. E. True, Jensen Hughes, Inc.
D. J. Wakefield, ABS Consulting, Inc.
I. B. Wall, Individual
T. A. Wheeler, Sandia National Laboratories
J. W. Young, GE Hitachi Nuclear Energy

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:

K. Woodard, *Chair*, ABS Consulting, Inc. (retired)
G. A. Teagarden, *Vice Chair,* Jensen Hughes, Inc.

N. E. Bixler, Sandia National Laboratories
A. R. Caldwell, *Associate Member*, Lloyd's Register Consulting
K. Compton, U.S. Nuclear Regulatory Commission
D. H. Johnson, ABS Consulting, Inc.
G. W. Kindred, Tennessee Valley Authority
S. H. Levinson, AREVA Inc.
C. A. Mazzola, Chicago Bridge & Iron Federal Services
J. A. Mitchell, U.S. Nuclear Regulatory Commission
V. Mubayi, Brookhaven National Laboratory (retired)
K. O'Kula, AECOM
P. D. Paul, Duke Energy

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."

JCNRM Subcommittee on Standards Development

B. D. Sloane, *Chair*, Jensen Hughes, Inc.
D. W. Henneke, *Vice Chair*, General Electric Company
(Alternate: Y. J. Li, GE Hitachi Nuclear Energy)

V. K. Anderson, Nuclear Energy Institute
S. Bernsen, Individual
J. H. Bickel, Evergreen Safety & Reliability Technologies, LLC
E. T. Burns, Jensen Hughes, Inc.

J. R. Chapman, Scientech, Inc.
H. L. Detar, Westinghouse Electric Company LLC
(Alternate: **N. Larson,** Westinghouse Electric Company LLC)
M. Drouin, U.S. Nuclear Regulatory Commission
(Alternate: **C. J. Fong,** U.S. Nuclear Regulatory Commission)
K. N. Fleming, KNF Consulting Services, LLC
E. A Hughes, Etranco, Inc.
M. T. Leonard, Individual
S. R. Lewis, Electric Power Research Institute
Z. Ma, Idaho National Laboratory
J. O'Brien, U.S. Department of Energy
V. Patel, Southern Nuclear Operating Company
M. Sattison, Idaho National Laboratory
V. Sorel, Électricité de France
S. D. Unwin, Pacific Northwest National Laboratory
D. J. Wakefield, ABS Consulting, Inc.
T. A. Wheeler, Sandia National Laboratories
K. Woodard, ABS Consulting, Inc. (retired)
F. Yilmaz, South Texas Nuclear Operating Company

JCNRM Subcommittee on Standards Maintenance

P. J. Amico, *Chair*, Jensen Hughes, Inc.
A. Maioli, *Vice Chair*, Westinghouse Electric Company LLC
G. W. Parry, *Vice Chair*, Jensen Hughes, Inc.

V. Andersen, Jensen Hughes, Inc.
J. H. Bickel, Evergreen Safety & Reliability Technologies, LLC
J. M. Biersdorf, Idaho National Laboratory
R. J. Budnitz, Lawrence Berkeley National Laboratory
M. P. Carr, Curtiss-Wright/Scientech, Inc.
M. R. Denman, Sandia National Laboratories
K. R. Fine, FirstEnergy Nuclear Operating Company
H. A. Hackerott, Omaha Public Power District–Nuclear Energy Division
J. Hall, Entergy Operations, Inc.
D. C Hance, Electric Power Research Institute
D. G. Harrison, U.S. Nuclear Regulatory Commission
(Alternate: **C. J. Fong,** U.S. Nuclear Regulatory Commission)
T. G. Hook, Arizona Public Service
E. A Hughes, Etranco, Inc.
A. M. Kammerer, Annie Kammerer Consulting, LLC
K. L. Kiper, Westinghouse Electric Company LLC
S. Kojima, Kojima Risk Institute, Inc.
J. C. Lin, ABS Consulting, Inc.
D. N. Miskiewicz, Engineering Planning and Management, Inc.
P. F. Nelson, National Autonomous University of Mexico
S. P. Nowlen, Sandia National Laboratories

M. K. Ravindra, MKRavindra Consulting
A. A. Rubbicco, Westinghouse Electric Company LLC
R. E. Schneider, Westinghouse Electric Company LLC
R. Sewell, R. T. Sewell Associates
K. Sutton, INGRID Consulting Services, LLC
M. L. Szoke, Individual
I. B. Wall, Individual
J. W. Young, GE Hitachi Nuclear Energy

JCNRM Subcommittee on Risk Application

G. W. Kindred, *Chair*, Tennessee Valley Authority
G. M. Demoss, *Vice Chair*, PSEG Nuclear LLC
D. M. Jones, *Vice Chair*, Maracor, A Division of Enercon Services, Inc.

R. J. Budnitz, Lawrence Berkeley National Laboratory
J. M. Jansen Vehec, JTV Nuclear Consultants
K. L. Kiper, Westinghouse Electric Company LLC
S. H. Levinson, AREVA Inc.
L. A. Mrowca, U.S. Nuclear Regulatory Commission
P. F. Nelson, National Autonomous University of Mexico
P. J. O'Regan, Electric Power Research Institute
V. Patel, Southern Nuclear Operating Company
K. Sutton, INGRID Consulting Services, LLC
C. Trull, Westinghouse Electric Company LLC

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

Section 1
Introduction

1.1 OBJECTIVE

This Standard sets forth requirements for the consequence analysis portion of probabilistic risk assessments (PRAs) used to support risk-informed decisions for accidents involving the release of radioactive materials into the atmosphere. It is expected that the primary use of this Standard would be in support of nuclear power plants, although it could support broader applications. In these cases, supplemental requirements may be needed to ensure technical adequacy. This portion of a PRA is typically known as a Level 3 analysis.

1.2 COORDINATION WITH OTHER PROBABILISTIC RISK ASSESSMENT STANDARDS

This Standard was developed based on the body of knowledge and experience accumulated through the development and application of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (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," [1] and the 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)," [2] 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.

1.3 PURPOSE AND SCOPE

Consequence analysis assesses the effect of releases of radionuclides on the surrounding population and the environment. This Standard only includes limited treatment of the impact on doses of the release of radioactive materials that could reach liquid pathways (i.e., due to deposition onto land and bodies of water).

To date, there have been few consequence assessments dealing with liquid releases from nuclear facilities. Such releases would include releases in liquid form into rivers, lakes, estuaries, and oceans. In addition, releases could reach aquifers via transport through geological media. The rationale for not treating liquid releases in consequence analyses has typically been due to adequate time available for interdiction of foodstuffs and relocation. Therefore, this Standard does not address transport through geological media and into aquifers or releases of radioactive material directly into surface water bodies.

Consequence modeling can therefore be defined as a set of calculations of the ranges of potential adverse impacts (in terms of probabilities of occurrence and magnitudes) that would follow from the dose received by humans due to a release of radionuclides. These adverse impacts, commonly referred to as "public risks," include (1) early fatalities, (2) latent cancer fatalities, (3) early injuries, and (4) non-fatal cancers. In addition, adverse impacts can occur due to contamination of property, land, and surface water. Consequence analyses may include assessments of the economic impact of dose avoidance strategies, such as relocation of population, land and structure decontamination, and interdiction of foodstuffs.

Consequence modeling provides the means for relating these risks to the characteristics of the radioactive release and has many actual or potential applications including the following examples:

(a) risk evaluation, generic or site-specific, individual or the general population
(b) environmental impact assessment
(c) rulemaking and regulatory procedures
(d) emergency response
(e) development of criteria for the acceptability of risk
(f) instrumentation needs and dose assessment
(g) facility siting
(h) comparison with safety goals evaluation
(i) evaluation of alternative design features (e.g., severe accident mitigation alternatives (SAMAs) analysis)
(j) cost-benefit analyses

A Level 3 analysis incorporates information including demography, emergency planning, physical properties of radionuclides, meteorology, atmospheric dispersion and transport, size of nearby structures, health physics, and other disciplines. Use of this information is detailed in this Standard.

While the primary use of this Level 3 PRA Standard is most likely to be for LWRs, the methodology is generally applicable to any type of radioactive material released to the atmosphere for which the release characteristics can be defined. It is recognized, however, that there may be specific applications where the source term phenomenology and atmospheric dispersion are complex. Examples of potential analyses may include

(a) releases of dense and/or reactive gases (e.g., UF6) that can have complex release and transport characteristics;
(b) releases of tritium or carbon-14, which behave differently in the environment (e.g., deposition followed by re-emission); or
(c) energetic releases (i.e., explosions where momentum effects might be significant).

Although there may be available analytical tools for determining such consequences, the Supporting Requirements (SRs) in this Standard may not fully address such phenomenology. Section 3 of this Standard outlines a process by which the completeness of the requirements is assessed and supplemented to meet analytical requirements. This includes the selection of appropriate models. Additionally, Section 7 of this Standard provides peer review requirements to ensure technical adequacy of the analysis.

1.4 STRUCTURE FOR LEVEL 3 REQUIREMENTS

1.4.1 Level 3 Technical Elements

The technical requirements for the Level 3 analysis are organized by their respective technical elements. These technical elements define the scope of a Level 3 analysis. Sections 4 and 5 discuss these technical elements in detail.

1.4.2 High Level Requirements

A set of objectives and high level requirements (HLRs) is provided for each technical element in the Technical Requirements (Section 4 of this Standard). The HLRs set forth the minimum requirements to assess the technical adequacy of a Level 3 analysis, independent of an application. The HLRs are defined in general terms and present the top-level logic for the derivation of more detailed SRs.

1.4.3 Supporting Requirements (SRs)

A set of SRs is provided for each HLR in Sections 4 and 5. Multiple HLRs are defined for each technical element.

This Standard is intended to support a wide range of applications that require a corresponding range of Level 3 analysis capabilities. Applications vary with respect to which risk metrics are employed, which decision criteria are used, the extent of reliance on the results to support a decision, and the degree of resolution required for the factors that determine the risk significance of the subject of the decision. In developing the different portions of the Level 3 PRA model, it is recognized that not every technical element (e.g., atmospheric transport and dispersion model) will be or needs to be developed to the same degree of site specificity or the same degree of realism.

1.4.4 Capability Categories

The types of risk-informed PRA applications contemplated under this Standard are very broad and include applications related to design, emergency response, meteorological programs, licensing, and many other disciplines. Both regulatory risk-informed applications and applications not involving U.S. Nuclear Regulatory Commission (NRC) regulations are contemplated.

Although the range of capabilities required for each portion of the PRA to support an application falls on a continuum, three levels are defined and labeled either Capability Category I, II, or III, so that requirements can be developed and presented in a manageable way. Table 1-1 describes, for two principal attributes of PRA, the bases for defining the Capability Categories. This table was used to develop the SRs for each HLR. It is noted that Table 1-1 in this Standard excludes the attribute of scope and level of detail associated with plant design, operation, and maintenance used in the analogous table in the ASME/ANS PRA Standard (RA-Sb-2013 [1]), because this attribute is not generally applicable to Level 3 analyses. The two attributes of site specificity and realism provide adequate means to differentiate Capability Categories.

The intent of the delineation of the Capability Categories within the SRs is generally that the degree of site specificity and the degree of realism increases from Capability Category I to Capability Category III. However, the Capability Categories are not based on the level of conservatism (i.e., tendency to overestimate risk due to simplifications in the PRA) in a particular aspect of the analysis. The level of conservatism may decrease as the Capability Category increases and more detail and more realism are introduced into the analysis. However, this is not true for all requirements, and this should not be assumed.