Chapter 2
Standards for Probabilistic Risk/Safety Assessments of Nuclear Power Plants and High-Level Nuclear Safety Goals—An Overview

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Introduction

Probabilistic Safety/Risk Assessment (PSA/PRA) is a widely accepted technology for investigating the risks posed by hazardous facilities. In the case of commercial nuclear power plants, the use of PSA/PRA for exploring and estimating risk goes back almost 50 years to 1975 when the pioneering Reactor Safety Study (RSS) known as WASH-1400 (US NRC 1975) carried out a quantitative risk evaluation of light-water reactors (LWRs) in the US, including pressurized water reactors (PWRs) and boiling water reactors (BWRs). The insights into the multifarious factors influencing plant risk that were revealed in the RSS helped to bring risk assessment technology closer to the regulatory arena. The results and analyses of WASH-1400 were considerably enhanced by the study of severe accident risks in PWRs and BWRs with different containment designs (large volume containments, pressure suppression containments, and ice condenser containments) carried out by the US Nuclear Regulatory Commission (NRC) in 1990 (US NRC 1990).

In 1995, the US Nuclear Regulatory Commission issued the PRA Policy Statement (US NRC 1995) that stated:

The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data, and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

The objective of the PRA policy statement was to ensure that the applications of PRA in the nuclear industry were implemented in a consistent and predictable manner that would promote the stability and efficiency of regulatory decisions. In

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P. V. Varde et al. (eds.), *Advances in Risk-Informed Technologies*, Risk, Reliability and Safety Engineering, https://doi.org/10.1007/978-981-99-9122-8_2

addition, the policy statement directed that the agency should use PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce risks wherever possible as well as unnecessary conservatism that was associated with deterministic regulatory requirements.

Development of PRA Standards

The increased use of PRAs/PSAs by the industry as well as in the regulatory decision-making process required that PRAs/PSAs be formulated and conducted in an accurate and consistent manner. This meant that that the quality, scope, methodology, and data used in PRAs/PSAs meet certain minimum performance standards. To achieve this objective, professional societies, industry, and the staff have undertaken initiatives to develop national consensus standards and guidance on the use of PRA in regulatory decision-making (US NRC 2020). 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 for commercial nuclear power plants that would be acceptable to the nuclear utilities, and the regulator, NRC, and structured around the three Levels of PRA for LWRs (i.e., Level 1, Level 2, Level 3) to be jointly issued by the two societies.

A PRA of an LWR is conventionally divided into three phases: Level 1 PRA carries out the analysis of the accident from the initiating event until the onset of core damage. Level 2 PRA focuses on the probabilistic treatment of accident progression from the release of core fission products to their transport in the containment and potentially from the containment to the environment. Level 3 PRA analyses the atmospheric transport of the released plume beyond the plant boundary, depletion, and the potential radiation exposure of offsite individuals through different pathways such as cloudshine. groundshine, inhalation, etc., and their health impacts such as early fatalities or latent cancers taking protective actions like evacuation and sheltering into account. Level 3 PRA also calculates the potential economic impacts of accident releases due to possible long-term relocation of affected populations as well as decontamination and/or interdiction of land and property.

The Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications (ASME 2022), has been approved and published. (Large Early Release Frequency, LERF, is the frequency of severe, unmitigated, accident releases that occur early in a time frame before protective actions of the offsite population like evacuation can be effectively implemented so there is a potential for early health effects.)

The Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs) was approved earlier for trial use and pilot application (ASME 2014). A final version is expected to be published in the near future.

The Standard for Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications was also approved earlier for trial use and pilot application (ASME 2017). Revisions to the standard are currently ongoing and a final version is expected to be published in the near future.

National consensus PRA standards are aimed at providing a set of minimum requirements that should be met, for a PRA of a plant to be considered acceptable. These standards include technical requirements of the various elements of a PRA that are focused on *what* is needed to perform that element in an acceptable way rather than *how to* perform that element. The process-related requirements address maintenance, upgrades and peer review of the standard and PRA configuration control. The peer review determines whether a PRA meets the requirements of the PRA standard.

Level 3 PRA Standard

The following material is based on the draft Level 3 PRA standard (ASME 2017) and indicates its scope and content.

The Level 3 Standard sets forth requirements for the probabilistic consequence analysis portion of PRAs/PSAs used to support risk-informed decisions for accidents involving the release of radioactive materials into the atmosphere. This portion of a PRA/PSA is typically known as a Level 3 analysis. This Standard also sets forth requirements for risk estimation 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 Level 3 consequence analysis for a Level 3 PRA.

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."

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

Probabilistic consequence modeling can be defined as a set of calculations of the ranges of potential adverse impacts (i.e., risk, 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 also include assessments of the economic impact of dose avoidance strategies, such as relocation of population, land and property decontamination, and interdiction of foodstuffs.

Probabilistic 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 facility-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), and
(j) cost–benefit analyses.

This Level 3 PRA Standard supports the quantification of a wide range of consequences/impacts to the public and the environment in the form of conditional consequences, given the postulated release(s) and risk when conditional consequence results are combined with the frequency results of the Level 1 PRA and the

Level 2 PRA. Examples of the potential Level 3 consequence metrics include:

- Total effective dose (e.g., individual at the site boundary, population dose within a specified distance)
- Specific organ dose (e.g., thyroid, lung)
- Early health effects (e.g., radiological injuries, radiological fatalities)
- Latent health effects (e.g., cancers)
- Land contamination levels (e.g., area exceeding a specific Cs-137 activity)
- Economic impacts (e.g., evacuation costs, economic disruption costs, remediation costs)
- Risk values when individual consequence results are combined with release frequencies (e.g., early individual fatality risk and LCF risk for comparison to the NRC QHOs).

While the primary use of the 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 from other nuclear facilities such as research reactors and fuel cycle facilities for which the release characteristics can be defined. There may be specific facilities and applications, however, where the source term phenomenology and atmospheric dispersion are complex such as releases of dense and/or reactive gases (e.g., uranium hexafluoride) that can have complex release and transport characteristics. In these cases, supplemental requirements may be needed to ensure technical acceptability.

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

This Standard contains the requirements for the following technical elements of a consequence analysis: (1) radionuclide release characterization; (2) protective actions; (3) meteorological data; (4) atmospheric transport and dispersion; (5) dosimetry; (6) health effects; (7) economic factors; (8) conditional consequence quantification; and (9) risk estimation.

The requirements of each technical element may be defined at two different levels: (1) high-level requirements (HLRs), that capture the overall objective of each element and (2) supporting requirements (SRs) that are defined for each HLR and are the minimal requirements needed to satisfy the HLR The HLRs generally address attributes of the PRA technical elements, such as (a) scope and level of detail, (b) model fidelity and realism, and (c) output or quantitative results.

Objectives were established for each technical element used to characterize the respective scope of a consequence analysis. The objectives reflect substantial experience accumulated with consequence assessment development and usage. These objectives form the basis for development of the HLRs for each technical element, which were used in turn to define the SRs. The SRs are generally divided into two Capability Categories (CC), CCI and CC II. The intent of the delineation of the Capability Categories within the SRs is generally that the degree of facility specificity and the degree of realism increases from CC I to CC II. The choice of which CC to use and which SRs apply in a specific portion of a PRA depends on the application and needs a professional judgment of the analyst.

Nine technical elements are addressed in the Level 3 PRA standard. A very brief summary description of HLRs and SRs is appended under each technical element below. Detailed descriptions of the objectives, the HLRs and the SRs under each CC for each technical element are provided in Reference 7.

(a) radionuclide release characterization for Level 3 analysis (RE)

The characteristics of the radionuclide release come from the Level 1 and mainly Level 2 analyses. They include: the release categories and their binning, the specifics of the source term of each release category (i.e., the quantity of various radionuclides released, the energy and height of the release and the timing of the release), the warning time for each release category, and the aerosol particle size of each release. The frequencies of the release categories come from the Level 1 and Level 2 analyses.

(b) protective action parameters and facility and regional data (PA)

The modeling of protective actions of the offsite population is based on criteria appropriate to the phase of the accident: in the early (or emergency) phase—the first hours or days of an incident—decisions regarding evacuation and/or sheltering of the offsite population are made and implemented based on facility status and anticipated or in-progress releases, in the intermediate phase–weeks to months after release—protective actions are based on environmental measurements of contamination, and in the late/long-term phase—the subsequent months to years following a release-recovery/remediation actions such as land/property decontamination are conducted and completed, and land/property is either released for unrestricted use or condemned for habitation. Parameters such as evacuation speed and shielding factors need to be based on site-specific studies of evacuation time estimates and building stock materials, and foodstuff interdiction criteria as well as contaminated land habitability criteria on recommendations of recognized official agencies. Other facility and regional data needed include facility physical characteristics (building dimensions, stack heights). Population distributions around the facility and the region, and regional land use data (fraction of land that is water, fraction that is farmland, agricultural production).

(c) meteorological data (ME)

Accurate meteorological data at and in the vicinity of the facility are important and should cover at least 90% of hourly data over a period of a year on windspeed, wind direction, precipitation and measurement of temperature difference with height or wind direction standard deviation or observations from representative weather stations that can be used to determine the atmospheric stability class. Met data should be collected under a qualified scheme of calibration, maintenance activities, and instrument exposures. Data from recognized sources on seasonal morning and afternoon mixing heights in the region need to be compiled.

(d) atmospheric transport and dispersion (AD)

The objectives of the atmospheric transport and dispersion technical element are to develop, use, and document an atmospheric transport and dispersion (ATD) model in such a way that: conditions at the facility are represented, site specific meteorological data is used, facility and accident specific attributes are accounted for, temporal and spatial changes in meteorological conditions are considered, and deposition of radionuclide particles is included. The requirement is to ensure that an appropriate dispersion methodology is adopted that incorporates the meteorological data to determine the airborne concentration and ground deposition for input into dose models.

(e) dosimetry (DO)

The dosimetry technical element uses appropriate dose conversion factors along with the computed radionuclide concentrations and surface depositions to determine the doses received by the tissues and organs of interest due to exposure to radioactive material via each of the relevant dose pathways in such a way that applicable exposure pathways and protective action impacts are included, and appropriate dose conversion factors are used. The plume concentrations and deposition resulting from the ATD model are used to calculate doses over the exposure period(s). The analysis includes applicable exposure pathways including cloudshine, groundshine, skin deposition, skin absorption, inhalation and ingestion, and takes into account the effect of protective actions on received dose. The calculation of groundshine dose integrates the dose over the exposure time period(s), accounting for deposited materials both during and after plume passage. Acute and committed doses from modeled pathways are calculated to obtain effective dose and specific organ doses for which health effects are to be estimated.

(f) health effects (HE)

This technical element estimates the health effects of interest, such as early fatalities, early injuries, latent cancer fatalities and non-fatal cancers based on the computed doses and appropriate risk factors in such a way that health effect modeling accounts for both dose and dose rate, using parameter values from recognized sources.

(g) economic factors (EC)

This technical element ensures that the economic factors determined for the analysis use appropriate models and facility-specific and regional data in such a way that economic model parameters are clearly defined, and parameter estimates have an appropriate basis. Economic cost factors include: evacuation costs, evacuation costs, relocation costs including temporary unemployment, land value, depreciation, crop losses, decontamination costs, loss of use of offsite property, and public health costs (e.g., based on monetizing population dose).

(h) conditional consequence quantification and reporting (QT)

Conditional consequences include metrics of interest such as doses, early fatalities, latent cancers, costs, etc. that are identified and quantified in this technical element.

(i) risk estimation (RI)

This technical element identifies the risk metrics of interest such as population dose risk, early fatality risk, latent cancer risk, etc., and estimates them by combining the results of the Level 1, Level 2, and Level 3 analyses. It also describes how the combined risk results of a Level 1, Level 2, and Level 3 PRA can be presented.

This Standard is being developed by experts in various disciplines associated with consequence analysis such as severe accident source term modeling, meteorology, atmospheric transport, dosimetry, radiation health effects and risk estimation drawn from U.S. nuclear utilities, national laboratories, individual consultancies, and the U.S. NRC. A very limited amount of international participation has taken place in the process of developing the standard, but international users should be able to adapt the examples to their specific applications and regulatory requirements.

Safety Goals

The output of a Level 3 PRA is typically expressed through risk metrics such as the public health risks of early fatality and latent cancer fatality caused by exposure to the radionuclides released in a severe reactor accident. In 1986, the US NRC issued a Safety Goal Policy Statement that adopted probabilistic safety goals as rational objectives for the limits of severe accidents on public health risk (8). The safety goals that were adopted included two qualitative safety goals and two quantitative health objectives (QHOs). The qualitative goals expressed the Commission's expectation that members of the public residing in the vicinity of a nuclear power plant (NPP) should bear no significant additional risk from plant operation and that risks of generating electricity by a NPP should be comparable to or less than the risks of electric generation by other viable technologies. The two QHOs bear directly on the public health risks calculated in a level 3 PRA. QHO 1 refers to individual early fatality risk and states that the risk of an early fatality from a NPP accident to a biologically average individual (in terms of age and other risk factors) who resides within 1 mile of the site exclusion area boundary should be less than one-tenth of one percent of the sum of prompt fatality risks from all other accidents that the US population is generally exposed to. QHO 2 refers to individual latent cancer fatality risk and says the risk of a latent cancer fatality from a NPP accident to a biologically average individual within 10 miles of the site should be less than one-tenth of one percent of the sum of cancer fatality risks to the US population from all other causes. The numerical value of one-tenth of one percent of the background risk of either individual early fatality or latent cancer fatality represented, in the Commission's view, the notion of no significant additional risk from NPP accidents.

The results of major Level 3 PRA risk studies such as the NUREG-1150 study referred to in Reference 2 indicated that nuclear power plants in the United States satisfy the safety goal QHOs by a wide margin even taking into account uncertainties in the analysis. Two decades after NUREG-1150, the State-of-the-Art Reactor Consequence Analyses (SOARCA) study (US NRC 2012), performed by Sandia National Laboratories for the NRC, revisited the methods of analysis used in NUREG-1150 and repeated probabilistic risk assessments for two nuclear plants—Surry and Peach Bottom also evaluated in NUREG-1150—using updated consequence analysis tools and methodologies. While SOARCA did "not examine all scenarios typically considered in a probabilistic risk assessment," its results indicated that the QHOs were satisfied by even larger margins than in the NUREG-1150 study.

However, in the wake of the severe reactor accident that occurred at Fukushima, there has been a realization that individual fatality risks from radiation exposure do not constitute the only risk to which offsite populations are exposed. To protect the offsite public from exposure to the radiological materials released to the environment during a severe accident, various emergency protective action measures are employed, ranging from sheltering-in-place to evacuation followed by extended relocation, if necessary, and remediation or condemnation of contaminated land along with banning of contaminated food. At Fukushima, there were no radiation-induced early fatalities and any latent cancer fatalities from the radionuclides released in the accident are not expected to present a measurable increase over the background rate of cancer fatality in Japan. On the other hand, there were a significant number of non-radiation-related fatalities from traffic accidents during evacuation and some other causes such as premature deaths of relocated elderly patients. Other major consequences of the Fukushima accident have been the extended loss of homes and lands, the negative psychological impacts of long-term relocation, and the high costs involved with the remediation of contaminated land.

The non-radiation related risks of NPP accidents arise from the measures taken to prevent the offsite public from radiation exposure. A number of recent studies have suggested that rather than radiation-induced health effects the major impact of an NPP accident is what has been termed as societal risk, the social disruption caused by the relocation of large numbers of people. Bier et al. (2014) analyzed accidents at five sites in the US and concluded that the number of people relocated represents a viable measure of societal impact. Denning and Mubayi (2016) compared the monetized impact of NPP accidents to other societally disruptive events such as major hurricanes and observed that societal risk rather than individual health risk of radiation exposure is the dominant risk of NPP accidents.

Mubayi and Youngblood (2021) have proposed an additional safety goal based on societal risk to the existing NRC safety goals that consists of both a qualitative as well as a quantitative goal. The qualitative societal risk goal is that there should be no significant likelihood of extended relocation of a large number of people due to an NPP accident. The proposed quantitative goal is based on the recognition that it is the release of the relatively long-lived cesium isotopes in a NPP accident that is responsible for the extended relocation of the public in line with habitability criteria. Hence the proposed quantitative societal risk goal is that the mean frequency of release of cesium amounting to X% of the core inventory that can cause an extended relocation of more than one year of Y offsite persons should not exceed 1E-06 per reactor-year. The numerical values of X and Y are ultimately policy choices on the part of decision-makers. A preliminary review of the literature indicates that X% may range from 1 to −10% for most operating reactors in the US while the analysis in Ref. 10 suggests that Y may range from about 1E +04 to 1E+05 at several sites in the US.

A societal risk goal added to the existing US NRC safety goals would help to achieve a more comprehensive characterization of nuclear power plant safety.

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