Dealing with Beyond-Design-Basis Accidents in Nuclear Safety Decisions

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Abstract: This paper presents an approach for dealing with beyond-design-basis accidents in nuclear safety decisions. The proposed approach integrates traditional deterministic and risk considerations and is based on the following key principles: (a) limiting the radiological consequences of higher frequency accident sequences by defining an extended-design-basis accidents category or a design enhancement category within the traditional beyond-design-basis regime; (b) controlling the total risk by risk-informed cost beneficial enhancements to safety; and (c) consideration of uncertainty and decision stakes (e.g., consequences) in addressing the necessity and sufficiency that must be imposed on the application of defense in depth. The feasibility of implementing the proposed approach is also discussed in this paper.

Keywords: Beyond-design-basis accidents, Risk-informed enhancement to safety, Defense in depth

1. INTRODUCTION

Regulatory requirements for coping with abnormal events at a nuclear power plant can be categorized as those for anticipated operational occurrences (AOOs), those for design-basis accidents (DBAs), and those for Severe Accidents. AOOs are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit. Plants should be able to handle the full range of these AOOs with no fuel damage, and be returned to operation.

DBAs are more serious events that are not expected to occur during the life of a given plant. These postulated DBAs establish criteria for the design and evaluation of a variety of safety related systems and equipment. For DBAs, the possibility of limited damage to the fuel is accepted but off-site consequence limitations should not be exceeded.

A severe accident is a very low frequency event, brought about by multiple failures, which may result in changes to the reactor core configuration and significant radionuclide releases from the damaged core. In worst case severe accident scenarios, the reactor core becomes molten and the reactor containment is breached. These beyond-design-basis accidents are not usually analyzed in safety analysis reports. However, they are included in probabilistic risk assessment (PRA) studies. For severe accidents, historically, only a few direct regulatory requirements such as emergency planning were instituted. Severe accident regulatory decisions have mostly dealt with reducing the likelihood of such a serious accident rather than coping with one [1]. This approach was based on the assumption that, because of the "defense in depth" design philosophy, such accidents are of sufficiently low probability that mitigation of their consequences is not necessary for public safety.

[1] The views expressed in this paper are solely those of the author and do not necessarily represent those of either the ACRS or NRC.

The events at the Fukushima Dai-ichi Nuclear Power Station in Japan have provided an impetus for the re-examination of regulations for protection against severe accidents. The NRC established the Japan Near-Term Task Force (NTTF) to determine whether the agency should make additional improvements to its regulatory system. The NTTF found that the Commission's longstanding defense-in-depth philosophy, supported and modified as necessary by state-of-the-art PRA techniques, should continue to serve as the primary organizing principle of its regulatory framework. However, the Task Force concluded that the application of the defense-in-depth philosophy could be strengthened by including explicit requirements for beyond-design-basis events and recommended "establishing a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations" (Recommendation 1 of the NTTF) [2].

In early 2011, NRC Commissioner George Apostolakis led the Risk Management Task Force (RMTF) to develop a strategic vision and options for adopting a more comprehensive and holistic risk-informed, performance-based regulatory approach for reactors, materials, waste, fuel cycle, and transportation that would continue to ensure the safe and secure use of nuclear material. The RMTF report includes the recommendation that "the NRC should establish through rulemaking a design-enhancement category of regulatory treatment for beyond-design-basis accidents" [3].

In this paper an integrated approach for dealing with beyond-design-basis accidents is proposed. The proposed approach is consistent with the framework envisioned by the NTTF as well as with the RMTF recommendations for power reactors. The proposed approach integrates traditional deterministic and risk considerations and is based on the following key principles: (a) limiting the radiological consequences of higher frequency accident sequences by defining an extended-design-basis accidents category or a design enhancement category within the traditional beyond-design-basis regime; (b) controlling the total risk by risk-informed cost beneficial enhancements to safety; and (c) consideration of uncertainty and decision stakes (e.g., consequences) in addressing the necessity and sufficiency that must be imposed on the application of the traditional defense-in-depth philosophy. The feasibility of implementing the proposed approach is also discussed.

2. DETERMINISTIC AND PROBABILISTIC APPROACHES TO SAFETY ASSESSMENT

The principles of a traditional deterministic approach have been accepted by the NRC over many years to demonstrate the safety of design. The current regulatory framework is based largely, but not entirely on a deterministic approach that employs safety margins, operating experience, accident analyses, and a defense-in-depth philosophy. A deterministic approach specifies certain design and operational conditions and applies bounding criteria to demonstrate acceptable plant performance. Systems, structures, and components (SSCs) are designed and manufactured to accepted standards, regulations, codes of practice etc. to ensure that the SSCs can perform their intended functions.

Probabilistic Risk Assessment is being used increasingly as an important element in regulatory decision-making. In 1995, the NRC adopted a policy [4] that promotes increased use of probabilistic risk analysis in all regulatory matters to the extent supported by the state-of-the-art to complement the deterministic approach. The NRC has applied information gained from PRAs to complement other engineering analyses in improving issue specific safety regulation, and in changing the current licensing bases for individual plants. The NRC has made some revisions to its reactor regulations (10 CFR Part 50) to focus requirements on programs and activities that are most risk significant. However, these revisions provide alternatives that are strictly voluntary to current requirements.

Lack of coherence between probabilistic and traditional deterministic safety approaches has been identified as an issue for advancement of PRA technology in risk informed decision-making [5]. This was partly attributed to the lack of consistency between the accident sequences considered (predefined design basis accidents limited to single failures in active safety systems for deterministic evaluations vs. a systematic enumeration of accident sequences with all logical combinations of failures and successes of safety and non-safety systems in PRAs) as well as lack of a uniform assessment of acceptable risk of undesired consequences [5].

Figure 1 depicts an integrated representation of the technical analyses elements of both traditional deterministic and probabilistic approaches for informing the final safety decisions. In the following sections these elements together with the options for dealing with the beyond-design-basis severe accidents are discussed.

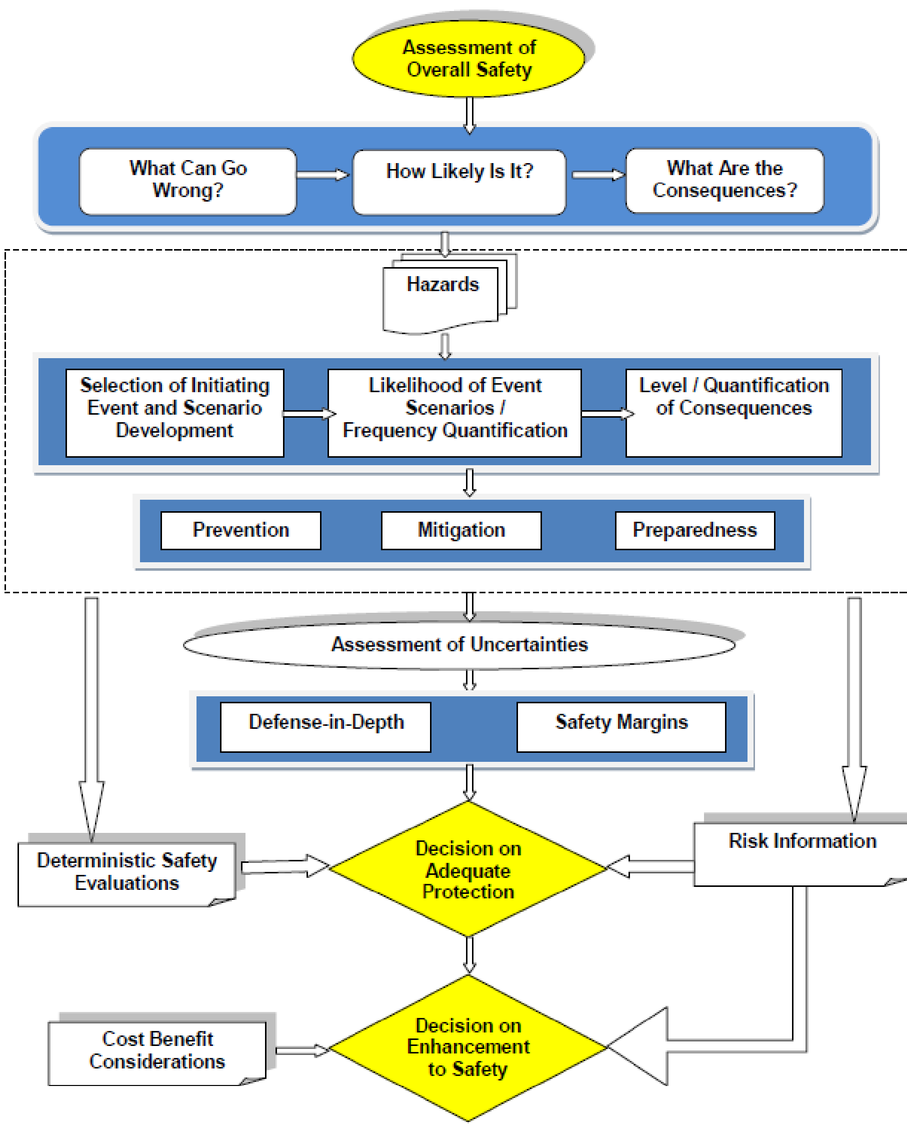

Figure 1 Integration of Technical Analyses Elements of Deterministic and Probabilistic Approaches for Informing the Final Safety Decisions

2.1 Identifying Events and Acceptance Criteria

All safety decisions are based, either explicitly or implicitly, on identifying radiological hazards and addressing the "risk triplet" [6] questions: "What can go wrong?"; "How likely is it?" ; and "What are the consequences?" The NRC addresses these three questions through the body of its regulations and guidance that it uses to regulate the many activities under its jurisdiction [7]. The deterministic approach assumes that adverse conditions can exist and establishes a specific set of design basis events (i.e., what can go wrong?). Some implied, but un-quantified, elements of probability are considered in the selection of the specific accidents to be analyzed as design basis events (i.e. how likely is it?). The deterministic approach then requires that the design include safety systems capable of preventing and/or mitigating the consequences (i.e., what are the consequences?) of those design basis events in order to protect public health and safety. The probabilistic approach considers all three questions in a more logical, explicit, and quantitative manner. PRA explicitly addresses a broad spectrum of initiating events and their event frequency. It then analyzes the consequences of those event scenarios and weights the consequences by the frequency, thus giving a measure of risk.

While the traditional deterministic approach to regulation has been successful in ensuring no undue risk to public health and safety, as suggested by both the RMTF and NTTF, the insights from the probabilistic approach can be used to strengthen the regulation by requiring measures to cope with additional events not foreseen in the design of the current operating plants.

There have been many efforts in the past to use PRA results for selecting the initiating events and categorizing the event sequences to be used as a basis for safety evaluation. The event sequences refer to a sequence of events starting from an initiating event challenging safety functions until a stable end-state is reached. The frequency-based categorization of event sequences in the past includes those proposed for pre-application safety evaluations of advanced reactor designs (i.e., PRISM and MHTGR) [8], Westinghouse Owners Group (WOG) risk-informed safety analysis approach [9], licensing basis event (LBE) selection for the pebble bed modular reactor (PBMR) [10], feasibility study for a risk-informed and performance-based technology-neutral regulatory structure for future plant licensing [11], and the Next Generation Nuclear Plant (NGNP) licensing basis event selection [12].

In its September 26, 2007 report on development of a technology-neutral regulatory framework, the ACRS concluded that, "the use of a frequency-consequence (F-C) curve is an appropriate way to establish a range of regulatory requirements to limit radiation exposure to the public." However, the Committee noted that, "a sequence-specific F-C curve, such as that developed in NUREG-1860, may not be a sufficient licensing criterion." The ACRS also noted that "a complementary cumulative distribution function (CCDF) F-C curve ("risk curve") that sums the contributions to risk from the entire spectrum of accident sequences establishes limits on risk better than the LBE F-C curve." The Committee was also concerned that "extension of the F-C curves to very low dose levels may unduly increase requirements for the scope and level of detail in the PRA performed to demonstrate compliance with the F-C curve" and may "detract attention from accidents which could have a more significant impact on public health and safety."

The traditional deterministic regulatory policies recognize the categorization of events and through explicit guidance and acceptance criteria constrain the allowable consequences of radiological releases from nuclear power plants. On the other hand, the Safety Goal Policy, which generally pertains to beyond- design-basis accidents, provides metrics in the context of individual and societal radiological risk.

The concept of a maximum credible accident is still used in the licensing process for the acceptability of a potential site (e.g. siting limits in 10 CFR 100) and performance requirements for the containment fission product cleanup systems and allowable leak rate (e.g.,10 CFR 50.34). Postulated is the fission product release into the containment associated with a substantial core-melt accident. Credible events have been interpreted in the past as events with frequencies higher than 10^{-6}. Therefore a coherent and logical way of dealing with beyond design basis severe accidents would be to limit the consequences of accidents with release frequencies higher than 10^{-6}. One option, for example, would be to have severe accident management/mitigation performance requirements to limit the consequences of such accidents consistent with the existing 10 CFR 50.34 dose limits at the exclusion area boundary (EAB).

2.2 Assessment of Uncertainties

Safety decisions whether risk-informed or not are made in the face of uncertainties and within the boundaries of the state of knowledge of nuclear power plants and how they behave under both normal and accident conditions [5]. Both deterministic and probabilistic safety evaluations must deal with uncertainties. However, the uncertainty should be examined in the context of a decision, focusing on the uncertainties that have impact on the outcome of the decision-making process.

Various classifications of uncertainty have been reported in the literature [13,14]. Two major groups of uncertainty that have been recognized are aleatory (or stochastic) and epistemic (or state-of-knowledge) uncertainty. The key distinction between these two types of uncertainty is that aleatory uncertainty is by definition irreducible. Epistemic uncertainty, on the other hand, can be reduced by further study.

There are two classes of epistemic uncertainty: parameter uncertainty and model uncertainty. Parameter uncertainties are those associated with the values of the fundamental parameters of a model, such as equipment failure rates that are used in quantifying the accident sequence frequencies in PRAs or the thermo-physical properties of the fuel, gap, and cladding that are used in quantifying the peak clad temperature (PCT) during a design-basis loss-of-coolant-accident (LOCA).

Model uncertainty reflects the limited ability to model accurately the specific events and phenomena. Examples include approaches to model human performance during accidents and models used for evaluating severe accident phenomena in Level 2 PRAs. Completeness, including possible “unknown unknowns,” can also be considered as one aspect of model uncertainty. Completeness uncertainty arises from the fact that not all contributors to risk are addressed in PRA models and not all phenomena and processes are addressed in deterministic evaluation models. Some contributors are not addressed because a methodology for their analyses has not yet been developed. For example, the influences of organizational performance cannot now be explicitly modeled in PRAs.

The safety philosophy of defense in depth and safety margins has been the traditional ways of dealing with uncertainties.

2.3 Defense-in-Depth

Historically, the term "defense in depth" has appeared frequently in the context of ensuing nuclear reactor safety. However, such term does not appear in Title 10 of the Code of Federal Regulations except in Appendix R of Part 50, where it appears once. Reference 15 provides some historical notes on defense-in-depth.

The practical implementation of defense in depth has often been associated with control of initiating event frequencies, redundancy and diversity in key safety functions, multiple physical barriers to fission-product release, and emergency response measures. This philosophy has been invoked primarily to compensate for uncertainty associated with the state of knowledge and understanding of the progression of accidents at nuclear power plants.

In its May 19, 1999 letter on the role of defense in depth in a risk-informed regulatory system, the ACRS noted that, "*defense in depth can still provide needed safety assurance in areas not treated or poorly treated by modern analyses or when results of the analyses are quite uncertain*." The Committee also recommended that "*to avoid conflict between the useful elements of defense in depth and the benefits that can be derived from quantitative risk assessment methods, constraints of necessity and sufficiency must be imposed on the application of defense in depth and these must somehow be related to the uncertainties associated with our ability to assess the risk*."

To address the issue of necessity and sufficiency that must be imposed on the application of defense in depth, an approach somewhat similar to the application of post-normal science [16] may be appropriate. The two key parameters of systems uncertainty and decision stakes (e.g., consequence) are considered (see Fig.2). Where these measures are low, the use of quantitative risk assessment techniques is seen as appropriate. However when the facts are uncertain, values in dispute, stakes high, and decisions urgent, the traditional defense in depth is seen as a remedy.

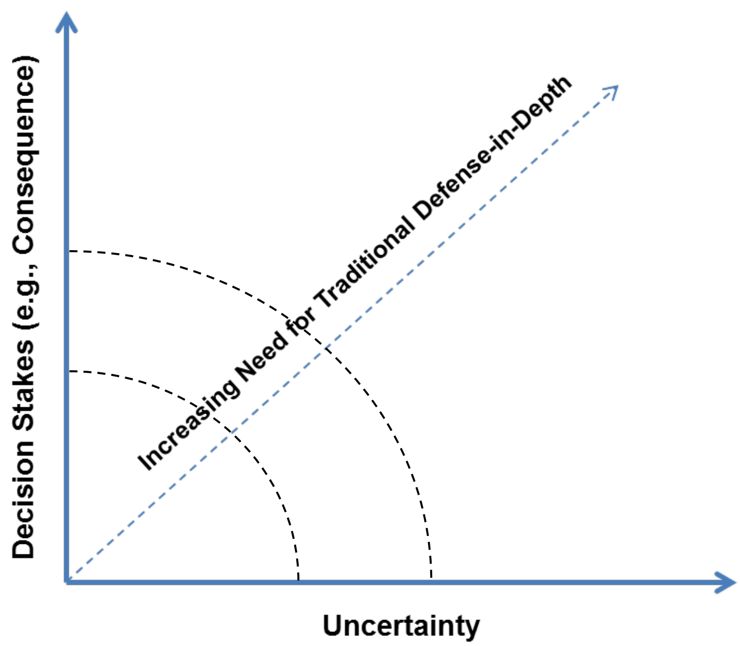

Figure 2. Application of defense in depth in relation to uncertainty and decision stakes

2.4 Safety Margins

The traditional deterministic regulatory approach also employs safety margins to ensure that the SSCs can perform their intended function. The safety margin (absolute term) is defined as the distance between an acceptance criterion (regulatory requirement) and a safety limit. The safety limit is a critical value of an assigned parameter associated with the failure of a system or a component. The licensing margin or safety margin (on the basis of analyses) is defined as the difference, in physical units, between an acceptance criterion and the results provided by either a best-estimate calculation or a conservative calculation. The most important safety margins relate to physical barriers against release of radioactive material. For LOCA analyses margin can be characterized as the difference between calculated parameters (e.g. peak fuel clad temperature, clad strain, maximum reactor coolant system pressure and stress, containment pressure and temperature, etc.) and the associated regulatory acceptance limit.

The LWR licensing approach has been historically based on the evaluation model (EM) methodology. This was established on the premise that deliberate thermal-hydraulic modeling conservatisms are included to compensate for lack of knowledge of the governing phenomena. This methodology was based on Appendix K of the U.S. Code of Federal Regulations (10 CFR Part 50). However, with improved understanding of the phenomena, there have been efforts to change the conservative biases and assumptions of the evaluation model methodology, allowing the licensee to move further toward best-estimate methodologies. Within the U.S. this led to a revision of the emergency core cooling system (ECCS) rule (10 CFR 50.46) in 1988 enabling licensees to apply best-estimate methodologies, with the provision that due allowance is given to any remaining uncertainties in code, data, or modeling.

2.5 Decision on Adequate Protection

In the U.S., by statute safety is measured by the standard of "adequate protection." In the context of nuclear safety, the terms "adequate protection" and "reasonable assurance of adequate protection" originated with the Atomic Energy Act of 1954 as amended. Section 182 of the Act gave the Atomic Energy Commission (AEC) broad authorities to establish “rule or regulation, deem necessary in order to enable it to find that the utilization or production of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public.”

The Atomic Energy Act of 1954 did not include a formal definition of "adequate protection" or "reasonable assurance of adequate protection." Rather, Congress left it up to the AEC to apply and give practical meaning to these terms. Today, the NRC operates under the same Congressional authorities. Historically the courts have been consistent in holding that defining “adequate protection” is left to the judgment of the NRC based on its technical expertise and on all the relevant information [17]. It is noted that the NRC requirements relating to the adequate protection of public health and safety do not consider costs.

2.6 Decision on Safety Enhancements

The Atomic Energy Act of 1954 (Section 161) also authorizes the NRC to establish by rule, regulation, or order, as the Commission “may deem necessary or desirable to promote the common defense and security or to protect health or to minimize danger to life or property.” This has been viewed by the Court of Appeals for the D.C. Circuit (Union of Concerned Scientists v. NRC, 824 F.2d 108, D.C. Cir. 1987) as a grant of authority to the Commission to provide a measure of safety above and beyond what is adequate protection, referred to by some as “safety enhancements.” The court also noted: “The exercise of this authority is entirely discretionary. If the Commission wishes to do so, it may order power plants already satisfying the standard of adequate protection to take additional safety precautions. When the Commission determines whether and to what extent to exercise this power, it may consider economic costs or any other factor. The Commission, after all, need not exercise the authority granted by section 161 at all; given this fact, the Commission certainly may use cost-benefit analysis to decide whether exercising the authority conferred by section 161 makes economic or policy sense.”

2.7 Cost-Benefit Considerations

Cost-benefit considerations are used as input to the NRC decisions whether to implement proposed regulatory actions. A regulatory analysis is performed to estimate benefits and costs, together with a conclusion as to whether the proposed regulatory action is "cost-beneficial" (i.e., benefits of the proposed action are equal to, or exceed, the costs of the proposed action). It should be noted that no legislation or regulation requires a regulatory analysis for NRC initiated actions. However, multiple Executive Orders have been issued on this topic over the past several years, and the NRC has been voluntarily performing such analyses since 1976. Cost benefit analysis is also used to support any backfit that represents an enhancement to safety beyond what may be required for adequate protection

3. AN INTEGRATED APPROACH FOR DEALING WITH BEYOND-DESIGN-BASIS ACCIDENTS

In this Section an integrated approach for dealing with beyond-design-basis accidents is discussed. The proposed approach is consistent with the framework envisioned by the NTTF as well as with the recommendations for power reactors in RMTF report. The feasibility of implementing the proposed approach is also discussed.

The proposed integrated approach integrates traditional deterministic and risk considerations and is based on the following key principles:

- limiting the radiological consequences of higher frequency accident sequences by defining an extended design-basis accidents category or a design enhancement category within the traditional beyond design basis regime,
- controlling the total risk by risk-informed cost beneficial enhancement to safety, and
- Consideration of uncertainty and decision stakes (e.g., consequences) in addressing the necessity and sufficiency that must be imposed on the application of the traditional defense in-depth philosophy.

3.1 Design-Enhancement Category of Events

Performing plant specific Level-3 PRAs is the ideal way of identifying the new design-enhancement events. However as a cost effective interim approach, insights from previous risk studies including NUREG-1150 Study [18], results of the SPAR models, together with lessons learned from Fukushima event could be used to identify higher frequency event scenarios with the potential for high radiological releases to environment (e.g., release frequency $>10^{-6}$) to be included in the design-enhancement category. It is noted that the recent State-of-the-Art Reactor Consequence Analyses (SOARCA) Project [19] also used a similar threshold to identify accident scenarios with the potential for higher consequences. This approach is similar to an alternative, "although not the favored approach of the RMTF," suggested by RMTF as "a transition to the more risk-informed, performance–based alternatives" [Alternative 1 in Appendix H to Reference 3].

A set of generic design-enhancement accident scenarios can be defined for each type of reactor and containment design (e.g., BWRs with Mark I, Mark II, and Mark III containments, PWRs with large dry, sub-atmospheric, and ice-condenser containments). Based on the results of the NUREG-1150 Study and experience with SOARCA, it is envisioned that each reactor and containment design would not have more than a few events in the design-enhancement category. An example of such events would be station black out (SBO) scenarios considered in developing the post-Fukushima requirements for mitigating strategies.

It is recognized that there are plant specific features that may influence the likelihood and the severity of specific events or phenomena during the progression of severe accidents. The documented insights from IPE [20] and IPEEE [21] programs can be very helpful in identifying plant specific vulnerabilities leading to unique and significant failure modes. Such information could be used for plant specific implementation of any performance-based requirements to limit radiological consequences of such events.

A review of the existing design-basis events could also be performed to determine whether some elements of the current design-basis events could be better addressed within the design-enhancement category as suggested by RMTF. One such example would be consideration of LOCAs with break sizes of larger than the transition break size (TBS), proposed for risk-informing 10 CFR 50.46, in the design-enhancement category.

As recommended by RMTF, the design-enhancement events should be recognized as a specific category of beyond-design-basis events. The objective is to define consistent regulatory treatments for such events in terms of analysis techniques, reporting, and other requirements. Such an approach would also provide a systematic framework for identifying existing voluntary industry initiatives which are most safety significant and would clarify the role of severe accident management guidelines (SAMGs) and potential use of FLEX equipment for providing the desired protection from beyond-design-accidents.

As it was discussed earlier (refer to Section 4.2), a coherent and logical acceptance criterion for the design-enhancement accidents would have to be a performance-based requirement consistent with the existing 10 CFR 50.34 dose limits at the EAB. The feasibility of implementing such limits or other alternatives should be further evaluated.

3.2 Necessity and Sufficiency of Defense in-Depth

To address the issue of necessity and sufficiency that must be imposed on the application of defense in depth, a conceptual framework for integrating risk-information and traditional defense-in-depth for dealing with design enhancement events is presented in figure 3. This figure is conceptually somewhat similar to SSCs classification in 10 CFR 50.69. When the uncertainties are relatively low and the consequence is high (region 1 in Figure 3), the risk-information is used for the decision of additional protection from the design-enhancement scenarios. However, the decision whether protection from a particular event is important for adequate protection or for safety enhancement would continue to be made on case-by-case basis.

When the uncertainties are high (e.g., facts are uncertain, values in dispute) and consequences are high (region 2 in Figure3), and decisions urgent, the traditional defense in-depth is seen as a remedy. This is consistent with the post-Fukushima decision by the Commission to address the uncertainties associated with beyond-design-basis external events by requiring additional defense-in depth measures for mitigating the consequence of such events.

If the uncertainties are relatively low and the decision stakes (change in consequences of acting) are low (Region 3 in Figure 3), risk-information could be used to relax some elements of defense-in-depth requirements for the current design-basis events. One such example would be consideration of LOCAs with break sizes larger than the transition break size (TBS), proposed for risk-informing 10 CFR 50.46, in the design-enhancement category that would have less stringent treatment requirements.

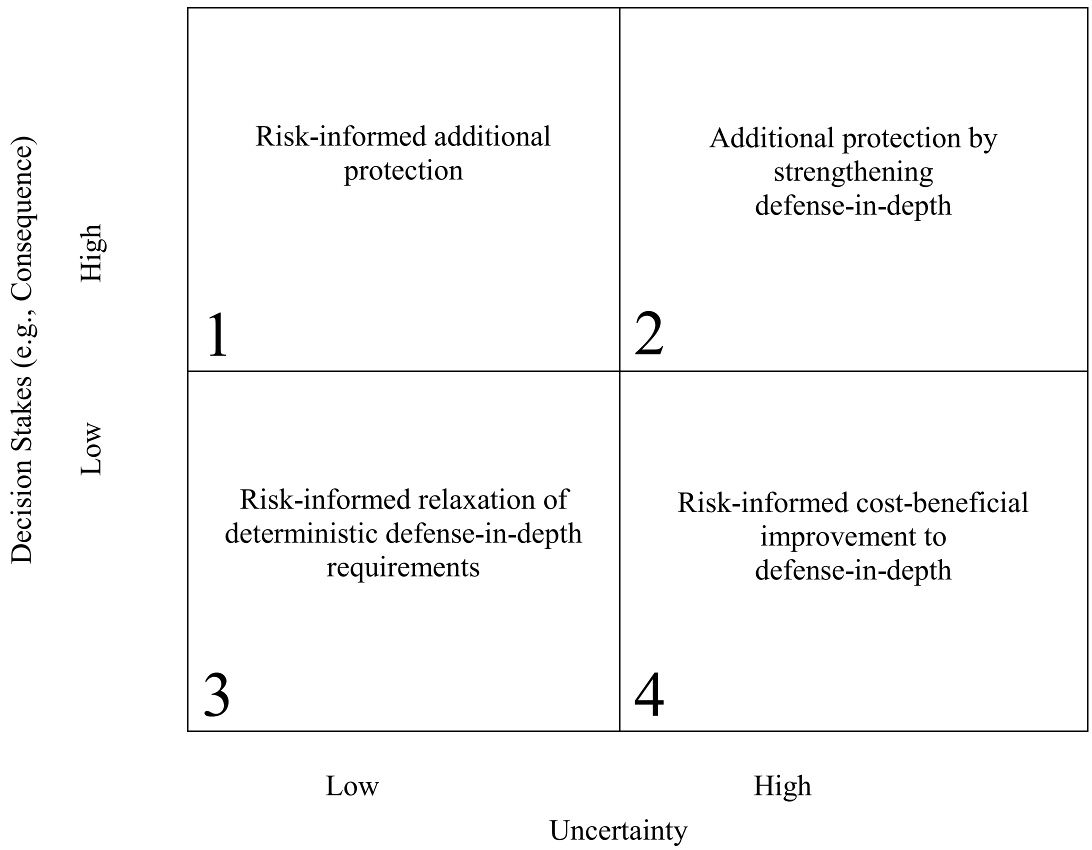

Figure 3 A Proposed Conceptual Framework for Integrating Risk-Information and Defense-in-Depth

There may also be circumstances that the uncertainties are high but the decision stake is low (region 4 in Figure 3). Under such circumstances the defense-in-depth measures could be improved if it is cost beneficial. This is consistent with the post-Fukushima decision requiring enhanced spent fuel pool instrumentation. It is noted that the Commission used an administrative exemption from the backfit rule for this requirement.

It is noted that risk information can be used for more quantitative characterization of defense-in-depth as suggested by the RMTF. Quantitative defense-in-depth importance measures should be developed to complement the risk importance measures for integrated risk-informed decision-making process. Such importance measures should provide metrics for the decisions of necessity or sufficiency of the defense-in-depth measures.

3.3 Risk-informed Cost-Beneficial Enhancement to Safety

As it was noted by the RMTF [3], “the determination of whether protection from a particular event is important for adequate protection or as a safety enhancement would continue to be made on case-by-case basis.” However, as noted before, cost benefit analysis is used to support any backfit that represents an enhancement to safety beyond what may be required for adequate protection. According to the Backfit Rule (10 CFR 50.109), such backfits may only be imposed if the NRC determines that “there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

A complete regulatory analysis will provide all the information necessary for backfit analysis. However, the backfitting decision criterion differs from the regulatory analysis decision criterion in that a "substantial increase" is needed to justify backfitting. In a June 30, 1993 Staff Requirement Memorandum (SRM), the Commission indicated that "substantial" means "important or significant in a large amount, extent, or degree." but the Commission has not set thresholds for a substantial increase. The Commission believed "these words embody a sound approach to the 'substantial increase' criterion and that this approach is flexible enough to allow for qualitative arguments that a given proposed rule would substantially increase safety" [22].

Several factors are considered to determine whether the backfit would provide a substantial increase in protection to public health and safety or common defense. For backfits associated with nuclear reactors, typically a safety goal screening evaluation is used as a surrogate for such a determination. According to the NRC's Regulatory Analysis Guidelines [23], "if the proposed safety goal screening criteria are satisfied, the NRC considers that the substantial additional protection standard is met for the proposed new requirement." Once it is decided that the potential backfit would result in a substantial increase in protection, it is then determined whether it is cost-justified in light of this increased protection.

The Regulatory Analysis Guidelines [23] proposes the use of the mean core damage frequency (CDF) and the conditional probability of early containment failure for the subsidiary safety goal screening criteria. Although CDF and large early release frequency (LERF) are appropriate safety goals subsidiaries, they may not adequately address protection against certain design-enhancement scenarios. For example the safety goal screening criteria described in Regulatory Analysis Guidelines cannot be used to address accident-mitigating initiatives that only improve the containment performance and have no impact on core damage frequency. As a part of the proposed integrated regulatory approach, the screening criteria for "substantial increase in the overall protection of the public health and safety" in regulatory analysis guidelines could be revised to also include criteria for strengthening the application of the defense-in-depth philosophy (e.g., figure 3 in conjunction with a threshold limit for frequency-consequences of the design-enhancement events).

4. SUMMARY AND CONCLUSIONS

An integrated approach for dealing with beyond-design-basis accidents was proposed in this paper. The proposed approach is consistent with the framework envisioned by the NTTF as well as with the recommendations of RMTF for power reactors. The proposed approach integrates traditional deterministic and risk considerations and is based on the following key principles: (a) limiting the radiological consequences of higher frequency accident sequences by defining an extended-design-basis accidents category or a design enhancement category within the traditional beyond-design-basis regime; (b) controlling the total risk by risk-informed cost beneficial enhancements to safety; and (c) Consideration of uncertainty and decision stakes (e.g., consequences) in addressing the necessity and sufficiency that must be imposed on the application of the traditional defense-in-depth philosophy. Such an approach would provide a systematic framework for identifying existing voluntary industry initiatives which are most safety significant and would clarify the role of severe accident management guidelines (SAMGs) and potential use of FLEX equipment for providing the desired protection from beyond-design-basis accidents.

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