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Severe Accident Regulatory Decisions: A Historical Perspective and the Role of the Advisory Committee on Reactor Safeguards

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Severe Accident Regulatory Decisions: A Historical Perspective and the Role of the Advisory Committee on Reactor Safeguards

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ABSTRACT

As part of its commitment to the U.S. Nuclear Regulatory Commission's knowledge management efforts, the Advisory Committee on Reactor Safeguards (ACRS) has begun an initiative to capture the Committee's institutional knowledge and memory. An important motivation for this initiative is to increase the effectiveness and efficiency of the Committee's review process by providing ready access to background information, insights, and understanding related to technical and regulatory issues. This report provides historical perspectives and insights on severe accident regulatory decisions. It also presents an overview of prior ACRS observations and recommendations regarding protection against severe accidents.

The views expressed in this report are solely those of the author and do not necessarily represent the views of the ACRS.

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ABBREVIATIONS

ABWR	advanced boiling-water reactor
ac	alternating current
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ALARA	as low as reasonably achievable
ALWR	advanced light-water reactor
AO	abnormal occurrence
AOO	anticipated operational occurrence
ARDC	advanced reactor design criteria
BDBE	beyond-design-basis event
BWR	boiling-water reactor
CFR	*Code of Federal Regulations*
DBA	design-basis accident
DBE	design-basis event
DOE	U.S. Department of Energy
DCH	direct containment heating
EAB	exclusion area boundary
EC-I	Event Category I
EC-II	Event Category II
EC-III	Event Category III
EC-IV	Event Category IV
EDMG	extensive damage mitigation guidelines
EOP	emergency operating procedure
EP	emergency preparedness
EPA	U.S. Environmental Protection Agency
ERDS	Emergency Response Data System
F-C	frequency-consequence
FEMA	Federal Emergency Management Agency
FLEX	diverse and flexible coping strategies
GDC	general design criterion/criteria
GL	generic letter
H_2	hydrogen
INL	Idaho National Laboratory
IPE	individual plant examination
IPEEE	Individual Plant Examination of External Events
ISG	interim staff guidance
JLD	Japan Lessons-Learned Division
LBE	licensing basis event
LF	limiting fault

LMP	Licensing Modernization Project
LOCA	loss-of-coolant accident
LWR	light-water reactor
MBDBE	mitigation of beyond-design-basis events
MHTGR	Modular High-Temperature Gas-Cooled Reactor
mrem	millirem
NEI	Nuclear Energy Institute
NEIMA	Nuclear Energy Innovation and Modernization Act
NEPA	National Environmental Policy Act
NGNP	Next Generation Nuclear Plant
NRC	U.S. Nuclear Regulatory Commission
NTTF	Near-Term Task Force
PAG	protective action guide
PBMR	Pebble Bed Modular Reactor
PRA	probabilistic risk assessment
PRISM	Power Reactor Innovative Small Modular
QHO	quantitative health objective
RCS	Reactor Coolant System
RES	Office of Nuclear Regulatory Research
ROP	Reactor Oversight Process
SAMA	severe accident mitigation alternative
SAMDA	severe accident mitigation design alternative
SAMG	severe accident management guideline
SARP	severe accident research program
SFP	spent fuel pool
SBO	station blackout
SRM	staff requirements memorandum
SSC	structure, system, and component
TID	technical information document
TMI-2	Three Mile Island, Unit 2
WOG	Westinghouse Owners Group
yr	year

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, as defined in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and categorized in Regulatory Guide 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," issued November 1978 (NRC, 1978), are those conditions of normal operation that are expected to occur one or more times during the life of a 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 it is required that offsite consequence limitations would not be exceeded.

A severe accident is a very low-frequency event, brought about by multiple failures, that results 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 typically addressed in safety analysis reports. However, they are included in probabilistic risk assessment (PRA) studies. Historically, only a few direct regulatory requirements for severe accidents, such as emergency planning, have been instituted. Severe accident regulatory decisions have mostly dealt with reducing the likelihood of such a serious accident, rather than coping with one. This approach assumed 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.

The 1979 accident at Three Mile Island Unit 2 (TMI-2) led to a reexamination of the design basis and consideration of regulations for protection against severe accidents. The first significant regulatory action for severe accident mitigation was the hydrogen rule (10 CFR 50.44, "Combustible gas control for nuclear power reactors") issued by the U.S. Nuclear Regulatory Commission (NRC) soon after the TMI-2 accident. This rule requires control of the hydrogen that is produced in a severe accident. Decisions were made to render the boiling-water reactor (BWR) Mark I and Mark II containments inert and install igniters for hydrogen control in BWR Mark III and ice-condenser containments.

The 1985 Commission policy statement on severe reactor accidents, regarding future designs and existing plants, affirmed the Commission's belief that a new design for a nuclear power plant can be shown to be acceptable for severe accident concerns if it meets certain criteria and procedural requirements, including "completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that may add to the assurance of no undue risk to public health and safety" (NRC, 1985).

In 1995, the NRC adopted a policy that promotes increasing the use of PRA in all regulatory matters, to the extent supported by the state of the art, to complement the deterministic approach (NRC, 1995a). The NRC has applied information gained from PRAs extensively to complement other engineering analyses in improving issue-specific safety regulation and in changing the current licensing bases for individual plants. The NRC has revised its reactor regulations to focus requirements on programs and activities that are most risk significant (Nourbakhsh, et al., 2018). The Advisory Committee on Reactor Safeguards (ACRS) has been very supportive of the evolution toward a risk-informed and performance-based regulatory system. The ACRS has taken a leading role in considering some of the challenges that have arisen in this effort.

The 2011 events at the Fukushima Dai-Ichi Nuclear Power Station in Japan provided an impetus for reexamining regulations for protection against severe accidents. Following this accident, the NRC required significant enhancements to U.S. nuclear power plants—including new equipment to better handle potential reactor core damage events—to ensure the nuclear industry and the NRC are prepared for the unexpected.

This report has been prepared as a part of the ACRS commitment to knowledge management and an initiative to capture the Committee's institutional knowledge and memory. An important motivation for this initiative is to increase the effectiveness and efficiency of the Committee's review process by providing ready access to background information, insights, and understanding related to technical and regulatory issues. This report provides historical perspectives and insights on severe accident regulatory decisions. It also presents an overview of prior ACRS observations and recommendations regarding protection against severe accidents.

2 EARLY CONSIDERATIONS OF SEVERE ACCIDENTS

The potential consequences of severe reactor accidents have been the subject of study since the earliest days of reactor development. The first estimates of consequences of severe accidents were published in the 1957 U.S. Atomic Energy Commission (AEC) report WASH-740, "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants" (AEC, 1957). This study attempted to provide the upper bounds of potential public hazards resulting from hypothetical severe accidents. Conservative values were used for many factors, influencing the magnitude of the estimated accident consequences. At the time, the technology and the state of knowledge about severe accidents had not progressed to the point where it was possible to use quantitative techniques to estimate the probability of such accidents. However, there was general agreement that the probability severe accidents in nuclear power reactors was exceedingly low.

Beginning in 1961, the AEC began defining a standard regulatory prescription for licensing. Reactor siting was the first issue addressed with the new approach. Regulations for site selection were developed as 10 CFR Part 100, "Reactor Site Criteria," in 1962. The regulation in 10 CFR Part 100 was developed, in part, based on assumptions that an upper limit of fission product release could be estimated and that the containment building, as a final independent line of defense against the release of radiation, would hold even if a serious accident took place. In conjunction with 10 CFR Part 100, the concept of a maximum credible accident was developed to evaluate the acceptability of a potential site (siting limits) and containment design requirements.

The regulation in 10 CFR Part 100 requires that the reactor site's suitability be judged, in part, based on a postulated fission product release (into the containment) associated with a "substantial meltdown" of the core. Currently licensed nuclear power plants relied on the characteristics of fission product release from the core into the containment set forth in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," both withdrawn in December 2016, and derived from the AEC report TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962 (DiNunno et al., 1962). TID-14844 assumed a core meltdown and instantaneous release of all noble gases, 50 percent of the iodine, and 1 percent of the other core particulate materials (solids) to the containment atmosphere.

The use of TID-14844 release assumptions has not been confined to a determination of site suitability. The regulatory applications of releases of this type cover a wide range, including the basis for (1) performance requirements of important fission-product cleanup systems such as sprays, (2) allowable containment leak rates, (3) the post-accident radiation environment for which the safety-related equipment should be qualified, (4) post-accident habitability requirements for the control room, and (5) post-accident sampling systems and accessibility.

By the mid-1960s, as the size of proposed plants increased significantly, the ACRS became concerned that a core meltdown accident, particularly one in which the plant's emergency core cooling system might fail to operate as designed, could lead to a breach of containment. Although the likelihood of such an event was considered to be extremely small, the potentially serious consequences were seen to justify careful study. In 1966, at the prodding of the ACRS, the AEC established a special task force to look into the problem of core meltdown (Walker and Wellock, 2010). The task force, chaired by former ACRS member William K. Ergen, issued its report in October 1967 (Ergen, 1967). The report offered assurances about the improbability of a core meltdown and the reliability of emergency core cooling system designs, but it also acknowledged that a loss-of-coolant accident (LOCA) could cause a breach of containment if the emergency core cooling system failed to perform. Therefore, containment could no longer be regarded as an unchallengeable barrier to the escape of radioactivity. This finding represented a "milestone in the evolution of reactor regulation" (Walker and Wellock, 2010).

In an ACRS letter on the task force's report, dated February 26, 1968, recognizing that absolute certainty cannot exist concerning any facet of safety, the Committee strongly recommended that a "positive approach be adopted toward studying the workability of protective measures to cope with core meltdown" (ACRS, 1968). The ACRS also stated that the task force's proposal—for the study of preventive measures to be made effective before loss of containment integrity to minimize the ultimate hazard—would be a helpful step in this direction. The Committee further recommended, as it did in its 1966 report on safety research, that a "vigorous program be aimed at gaining better understanding of the phenomena and mechanisms important to the course of large-scale core meltdown" (ACRS, 1966). The task force's report and ACRS recommendations formed the basis of some of the most important research initiatives and regulatory decisions by the AEC and the NRC, including the AEC's decision to undertake a study to estimate the probability of a severe accident, which resulted in the publication of the landmark Reactor Safety Study (WASH-1400) (NRC, 1975) and the beginning of the science of PRA as applied to nuclear power plant safety (NRC, 2003a).

3 PROTECTION AGAINST SEVERE ACCIDENTS FOLLOWING THE TMI-2 ACCIDENT

3.1 Lessons Learned from the TMI-2 Accident

The March 28, 1979, accident at TMI-2 led to a reexamination of the design basis and the consideration of regulations for protection against severe accidents. The reexamination of the design basis was prompted by the fact that the TMI-2 accident involved a small-break LOCA whose consequences should have been bounded by those of a large-break LOCA but became much more severe due to the operators' misunderstanding of the event. About half of the fuel melted before further progression of the accident was prevented.

Two weeks after the accident, President Jimmy Carter appointed a 12-member commission, headed by John Kemeny, then president of Dartmouth College, to investigate what had happened and its probable impact on the health and safety of the public and plant personnel. President Carter's Commission on the Accident at Three Mile Island (Kemeny Commission) issued its report in October 1979 (Kemeny et al., 1979), which contained several recommendations, such as the following:

> Continuing in-depth studies should be initiated on the probabilities and consequences (on-site and off-site) of nuclear power plant accidents, including the consequences of meltdown.
>
> Plans for protecting the public in the event of off-site radiation releases should be based on technical assessment of various classes of accidents that can take place at a given plant (Kemeny et al., 1979).

In May 1979, the NRC established a Lessons Learned Task Force to determine what actions were required for new operating licenses and chartered a Special Inquiry Group to examine all facets of the accident and its causes. The Special Inquiry Group, headed by attorney Mitchell Rogovin, reached many of the same conclusions as the Kemeny Commission. The January 1980 report by the Special Inquiry Group (Rogovin and Frampton, 1980) particularly states the following:

> …we have come far beyond the point at which the existing, stylized design basis accident review approach is sufficient. The process is not good enough to pinpoint many important design weaknesses or to address all the relevant design issues. Some important accidents are outside or are not adequately assessed within the 'design envelope'; key systems are not 'safety related'; and integration of human factors into the design review is grossly inadequate (Rogovin and Frampton, 1980).

The Lessons Learned Task Force led to the publication of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," issued July 1979 [NRC, 1979a], and NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report," issued October 1979 (NRC, 1979b). In its letter of December 13, 1979 (ACRS, 1979), on the TMI-2 Lessons Learned Task Force Final Report, the ACRS gave general support to many of the task force's recommendations. Regarding design features for core-damage and core-melt accidents, the Committee stated the following belief:

> …the recommendation should be augmented to require concurrent design studies by each licensee of possible hydrogen control and filtered venting systems which have the potential for mitigation of accidents involving large scale core damage or core melting, including an estimate of the cost, the possible schedule, and the potential for reduction in risk (ACRS, 1979).

The ACRS also made some comments and recommendations on several matters not directly addressed in NUREG-0578 or NUREG-0585, including the following:

> The lessons learned from the TMI accident should be viewed in a broader perspective. The Committee agreed that the TMI accident shows a need for considerable improvement in reactor and in knowledge of the behavior of plant operations during a wide range of transients. However, the Committee believed that there are other potentially important contributors to the probability of a reactor accident, and they should also receive priority attention.
>
> A re-evaluation should be made of the potential influence of a serious accident involving significant atmospheric release of radioactive materials from one unit of a multiple unit site on the ability to maintain the other units in a safe shutdown condition.
>
> The NRC Staff should give attention to the seismic implications of TMI, for example, the seismic qualifications of auxiliary feedwater supplies, the acceptability of failure of non-seismic Class 1 equipment, and the suitability of emergency procedures for earthquakes (ACRS 1979).

After the TMI-2 accident, the NRC decided that power reactor licensing should not continue until an assessment of the accident had been substantially completed and comprehensive improvements in both the operation and regulation of nuclear power plants had begun. About 9 months after the accident, the NRC proposed a post-TMI-2 action plan for utilities. In developing the action plan, the NRC assessed a range of recommendations and possible actions and rejected, adopted, or modified them. On June 16, 1980, the NRC issued its policy statement regarding the requirements to be met for current operating license applications (NRC, 1980a). The requirements were derived from NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," issued May 1980 (NRC, 1980b), and were documented in NUREG-0694, "TMI-Related Requirements for New Operating Licenses," issued June 1980 (NRC, 1980c).

In a letter of January 15, 1980 (ACRS, 1980a), in the draft NUREG-0660, the ACRS stated that the plan was comprehensive, but not selective:

> [This] comprehensiveness serves to dilute the items important to safety, and therefore important to termination of the licensing pause…. in the absence of priorities and identification of the items that the NRC Staff considers important, the ACRS finds it difficult to make objective comments on the Plan (ACRS, 1980a).

The Commission only approved specific items from NUREG-0660 for implementation at reactors. The NRC documented those items—including additional information about schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions—in NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980 (NRC, 1980d).

As a result of the degraded-core accident at TMI-2 and subsequent reevaluation of regulatory processes, the NRC published an advance notice of proposed rulemaking on October 2, 1980 (NRC, 1980s), announcing that it was considering amending its regulations to determine the extent to which commercial nuclear power plants should be designed to cope with reactor accidents beyond those considered in the current DBA approach. Principally, this rulemaking would have considered the need for nuclear power plant designs to be evaluated over a range of degraded-core cooling events with resulting core damage and a need for design improvements to cope with these events. In direct response to this advance notice of proposed rulemaking, the industry organized the Industry Degraded-Core Rulemaking program to provide an industry perspective for any rulemaking activities that might proceed. The NRC later withdrew the notice of proposed rulemaking (as discussed in section 3.7).

3.2 Hydrogen Rule

The first significant regulatory action for severe accident mitigation was the hydrogen rule (10 CFR 50.44), which the NRC issued after the TMI-2 accident. The rule required control of the hydrogen that is produced in a severe accident. Decisions were made to inert the BWR Mark I and Mark II containments and install igniters for hydrogen control in BWR Mark III and ice-condenser containments. Pressurized-water reactor (PWR) plants with large dry containments, (including those operating with a subatmospheric internal pressure) were exempted from hydrogen control because of the large volume of their containments.

3.3 Emergency Planning and Preparedness

It has long been recognized that emergencies could arise in the operation of reactor facilities. Reactor site criteria (10 CFR Part 100, published in 1962) state that a capability for taking protective measures on behalf of the public in the event of a serious accident should be established within a region called the low population zone surrounding a nuclear power plant site. In 1970, explicit requirements for plans to cope with emergencies were published in 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities".

In its letter of April 8, 1975, on emergency planning [ACRS, 1975], the Committee concluded that "an effective emergency plan can play a significant role in the protection of the nearby population in the unlikely event of a major accidental release of radioactive material for a nuclear installation." The Committee also stated the following:

> [S]sound emergency planning requires the ability to cope with a wide range of accident situations …. inquiries by the ACRS indicate a lack of development of an adequate series of scenarios to cover the range of emergencies which might take place and of methods for minimizing the resulting consequences (ACRS, 1975).

The Committee recommended that "such scenarios need to be developed and drills incorporating appropriate responses should be conducted" (ACRS, 1975).

In 1976, an ad hoc Task Force of the Conference of (State) Radiation Control Program Directors passed a resolution requesting the NRC to "make a determination of the most severe accident basis for which radiological emergency response plans should be developed by offsite agencies" (Collins et al., 1978). In November 1976, a task force of representatives from the NRC and the U.S. Environmental Protection Agency (EPA) was convened to address this request and related issues. The recommendations of the Task Force on Emergency Planning were published as NUREG-0396, EPA 520/1-78-016, "Planning Basis for the Development of

State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," issued December 1978 (Collins et al., 1978). The NRC/EPA Task Force did not attempt to define a single accident sequence or even a limited number of sequences. Rather, it identified the bounds of the parameters for which planning is recommended, based on knowledge of the timing, fission product release characteristics, and potential consequences of a spectrum of accidents. A few accident descriptions were considered, including the severe accident release categories of the Reactor Safety Study.

Although the TMI-2 accident was terminated without the need for a general evacuation, it made it clear that existing emergency planning requirements were unsatisfactory. The Commission requested immediate rulemaking on emergency planning. In December 1979, in accordance with the President's Commission on the Accident at Three Mile Island (Kemeny et al. 1979), the Federal Emergency Management Agency (FEMA) was designated as the lead agency for dealing with offsite nuclear power plant emergencies. In 1980, the NRC issued an emergency planning rule (10 CFR 50.47, "Emergency plans"), stipulating that the NRC would not issue a new operating license without a satisfactory emergency plan and that existing nuclear power plant owners had until April 1981 to develop an adequate emergency plan.

3.4 Source Term Reassessment

Following the publication of WASH-1400 and the TMI-2 accident, work was initiated to evaluate the predictive methods for calculating fission product release and transport. The results of this evaluation are contained in NUREG-0772, "Technical Bases for Estimating Fission Product Behavior during LWR Accidents," issued June 1981 (NRC, 1981a). The development of this report was prompted, in part, by a letter, dated December 21, 1980, from the Nuclear Safety Oversight Committee to President Carter, noting the questions raised at the time regarding iodine release and recommending that they should be answered by analyses and experimentation on an expedited basis. This evaluation resulted in several conclusions that represented significant departures from the assumptions in the Reactor Safety Study, including the conclusion that cesium iodide would be the expected predominant iodine chemical form under most postulated light-water reactor (LWR) accident conditions.

The potential impact of the NUREG-0772 findings on reactor regulation was also examined and the results documented in NUREG-0771, "Regulatory Impact of Nuclear Reactor Accident Source Term Assumption," issued June 1981 (NRC, 1981b). These studies formed the basis for designating five accident groups as representative of the spectrum of potential accident conditions.

In the 1980s, a substantial research program on severe accident phenomenology was initiated. Updated computational models for severe accident analysis were developed. A technical reassessment of severe accident source term technology for U.S. LWRs was published in NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms, Final Report," issued July 1986 (NRC, 1986a). This reassessment involved reviewing experimental and analytical results from severe accident research programs sponsored by the NRC and the nuclear industry.

The study documented in NUREG-1150 "Severe Accident Risks an Assessment for Five U.S. Nuclear Power Plants," issued December 1990 (NRC, 1990a), was a major effort to put the insights gained from the research on system behavior and phenomenological aspects of severe accidents into a risk perspective. An important characteristic of this study was its inclusion of uncertainties in the calculations of core damage frequency and risk due to incomplete understanding of reactor systems and severe accident phenomena.

The insights from the NUREG-1150 study have been used in several areas of reactor regulation, including the development of alternative radiological source terms for evaluating DBAs at nuclear reactors. In February 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (NRC, 1995b), which defined an alternative accident source term for regulatory applications. The magnitude and timing of radionuclide releases for the alternative accident source terms were derived from the insights of the NUREG-1150 source term analyses documented in NUREG/CR-5747, "Estimate of Radionuclide Release Characteristics into Containment Under Severe Accident Conditions," issued November 1993 (Nourbakhsh, 1993).

3.5 Safety Goal Policy Statement

In 1979, shortly after the accident at TMI-2, the ACRS recommended that the NRC consider establishing quantitative safety goals for nuclear power reactors. In its letter dated May 16, 1979, on quantitative safety goals (ACRS, 1979), the ACRS recognized the difficulties and uncertainties in quantifying risk and acknowledged that, in many situations, engineering judgment would be the only, or the primary, basis for a decision. Nevertheless, the Committee believed that the existence of quantitative safety goals and criteria could provide important yardsticks for such judgment.

The President's Commission on the Accident at Three Mile Island (Kemeny et al., 1979) and the NRC's Special Inquiry Group (Rogovin and Frampton, 1980) both recommended that the NRC better articulate its objectives and philosophy on the adequacy of reactor safety. In its response to the recommendations of the President's Commission, the NRC stated that it was "prepared to move forward with an explicit policy statement on safety philosophy and the role of safety-cost tradeoffs in the NRC safety decisions" (NRC, 1979c). The task of developing quantitative safety goals for nuclear power plants was just beginning.

The ACRS was at the forefront of the development of quantitative safety goals. The ACRS developed the first set of trial goals, published as *NUREG-0739,* "An Approach to Quantitative Safety Goals for Nuclear Power Plants," in October 1980 (ACRS, 1980b). These safety goals were the basis for the NRC's later work on the development of an NRC safety goal policy statement in 1986 (NRC, 1986b).

The NRC safety goal policy statement focuses on risks to the public from nuclear power plant operation. Its objective is to establish goals that broadly define an acceptable level of radiological risk. The Commission has established two qualitative safety goals that are supported by two quantitative objectives. These two supporting objectives are based on the principle that nuclear risks should not be a significant addition to other societal risks.

The qualitative safety goals are as follows:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following quantitative objectives are to be used in determining achievement of the above safety goals:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

Nourbakhsh (2021) offers a historical perspective on development of safety goals and implementation of the safety goal policy.

3.6 Backfit Rule

The AEC first adopted the backfit rule, 10 CFR 50.109, "Backfitting," in 1970. In justifying the need for a backfit rule, the AEC noted that "rapid changes in technology in the field of atomic energy result in the continual development of new or improved features designed to improve the safety of production and utilization facilities" (AEC, 1970). The purpose of the rule was to define "the circumstances under which the Commission may require backfitting of facilities—that is the addition or modification of structures, systems or components affecting the safety of the facility after construction permit has been issued".

The backfit rule was widely criticized for its lack of a systematic framework for the backfitting process. In particular, the President's Commission on the Accident at TMI noted that it "did not find the evidence that the need for improvement of older plants was systematically considered prior to Three Mile Island" (Kemeny et al., 1979).

In 1983, after concluding that the NRC regulations on backfitting and past staff practices did not adequately identify and justify proposed new requirements, the Commission issued a policy statement on backfitting and began rulemaking to revise the rule.

The Commission adopted a final backfit rule in 1985, but on appeal the U.S. Court of Appeals (*Union of Concerned Scientists v. NRC*, 824 F.2d 108, D.C. Cir. 1987) remanded that rule to the Commission because it failed to distinguish between "adequate protection" backfits for which the costs of the backfit could not be considered under the Atomic Energy Act, versus other backfits that represented an enhancement to safety beyond what might be required for adequate protection. The Commission subsequently adopted a revised backfit rule in 1988.

The backfit rule provides a disciplined process for the NRC to consider the imposition of new backfit requirements on licensees. According to 10 CFR 50.109, a backfit 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.

The rule also provides the exceptions that allow the NRC to impose a backfit regardless of cost.

3.7 Severe Accident Policy Statement

In the 1985 Commission policy statement on severe reactor accidents regarding future designs and existing plants (NRC, 1985), the Commission concluded, based on available information, that existing plants posed no undue risk to public health and safety and that there was no basis for immediate action on any regulatory requirements for those plants. At the same time, the Commission withdrew the October 2, 1980, advance notice of proposed rulemaking that invited public comment on long-term proposals for treating severe accident issues. However, based on NRC and industry experience with plant-specific PRAs, the Commission recognized that systematic examinations were beneficial in identifying plant-specific vulnerabilities to severe accidents that could be mitigated with low-cost improvements.

Regarding the decision process for certifying a new standard plant design—an approach the Commission strongly encouraged for future plants—the policy statement affirmed the Commission's belief that a new design for a nuclear power plant could be shown to be acceptable for severe accident concerns if it met certain criteria and procedural requirements, including the following:

> …completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that may add to the assurance of no undue risk to public health and safety (NRC, 1985).

3.8 Integration Plan for Closure of Severe Accident Issues

Subsequent to the issuance of the Severe Accident Policy Statement, the NRC was pursuing a number of separate programs on severe accidents. In 1988, the NRC coordinated these programs with an "Integration Plan for Closure of Severe Accident Issues" (NRC, 1988a). That plan consisted of six main elements: (1) individual plant examinations (IPEs), (2) containment performance improvements, (3) improved plant operations, (4) severe accident research (5) external events, and (6) accident management.

3.8.1 Individual Plant Locations

As a key part of the implementation of the Severe Accident Policy Statement, the NRC issued Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR 50.54(f)," on November 23, 1988 (NRC, 1988b), requesting that each licensee conduct an IPE "to identify any plant-specific vulnerabilities to severe accidents and report the results to the Commission." The purpose and scope of the IPE effort included examining internal events occurring at full power (including internal flooding but excluding internal fire). In response, the NRC received 75 IPEs covering 108 nuclear power plant units. The NRC examined the IPE submittals to determine what the collective IPE results implied about the safety of U.S. nuclear power plants and how the IPE program had affected reactor safety. The results of this review were documented in NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," issued December 1997 (NRC, 1997).

3.8.2 Containment Performance Improvements

The results of severe reactor research and risk assessments performed after the TMI-2 accident indicated relatively large uncertainties in the ability of LWR containments to successfully survive certain severe accident challenges. Based on this observation, the NRC concluded that certain generic severe accident challenges to each LWR containment type should be assessed to determine whether additional regulatory guidance or requirements concerning needed containment features were warranted, and to confirm the adequacy of existing Commission policy. The effort on containment performance improvements was integrated closely with the IPE program and was intended to focus on resolving hardware and procedural issues related to generic containment challenges (NRC, 1988a).

At the conclusion of the Mark I Containment Performance Improvement Program, several plant modifications that could substantially enhance the plants' capability to both prevent and mitigate the consequences of severe accidents were identified (NRC, 2011): (1) improved hardened Wetwell Vent capability, (2) improved reactor pressure vessel depressurization system reliability, (3) an alternative water supply to the reactor vessel and drywell sprays, and (4) updated emergency procedures and training. The NRC concluded that licensees evaluate the recommended safety improvements, with one exception (hardened Wetwell Vent capability), as part of the IPE Program. Additionally, the NRC issued GL 89-16, "Installation of a Hardened Wetwell Vent," dated September 1, 1989 (NRC, 1989a), indicating that it would approve hardened vents for licensees that proposed to install them and perform a backfit analysis for licensees that do not propose to install them.

Mark II containment vulnerabilities and potential improvements were similar to those identified for Mark I containments (NRC, 2011). However, less definitive conclusions were reached regarding the need for improved venting of Mark II containments. Aside from a change to Revision 4 of the Emergency Procedure Guidelines, titled "Steam Cooling," the staff did not identify any generic improvements that would be applicable to all Mark II containments. Therefore, the NRC requested that each licensee with a Mark II containment consider Mark I improvements, excluding the hardened vent, as part of its IPE.

For Mark III plants, potential improvements were also similar to those for Mark I plants (NRC, 2011). However, due to the relatively large volume of Mark III containment, the need for venting was found to be less likely than for the Mark I containment. In addition, some Mark III plants already had the capability to vent through a hardened system. A potential vulnerability for Mark III plants involved a station blackout (SBO), during which the igniters would be inoperable. Under these conditions, a detonable mixture of hydrogen (H_2) could develop, which could be ignited upon restoration of power. A potential improvement considered for Mark III containments was a backup power supply to enable the use of igniters during an SBO. However, no generic conclusions could be reached. Therefore, the NRC requested that each licensee with a Mark III containment consider the identified improvements as part of its IPE.

No generic improvements that would have been applicable to all ice-condenser containments were identified (NRC, 2011). The results of risk analysis for Sequoyah (a PWR with an ice-condenser containment) indicated that containment bypass sequences dominated early fatality risk. A separate NRC program on interfacing system LOCAs was underway to develop guidance and possible additional requirements for interfacing system LOCAs, including those that could bypass the containment. There was a great deal of uncertainty associated with the phenomenon of direct containment heating (DCH). Risk assessments varied considerably in their characterizations of DCH's contribution to containment failure.

As part of the Accident Management Program (discussed in section 3.8.6), full or partial depressurization of the reactor coolant system was being investigated as possible means to prevent or decrease the severity of DCH. An important finding was that depressurization to prevent DCH for ice-condenser plants was found not to be sufficient for preventing containment failure unless the igniters were operating to control the large amount of H_2 that could be produced. Containment failure resulting from uncontrolled H_2 burns or detonations was found to be a potentially important failure mode for ice-condenser containments (Nourbakhsh, 1990). This could occur in SBO events if power to the H_2 igniter system were lost, high concentrations of H_2 were produced because of core degradation, and power were then restored later. The NRC requested that each licensee with ice-condenser containment consider, as part of its IPE, the insights and improvements identified in the containment performance improvement program.

The NRC did not identify any generic improvements that would have been applicable to all dry containments. H_2 combustion on a global basis was not considered to be a significant threat to large, dry containments. However, less firm conclusions were reached for the smaller subatmospheric containments (NRC, 2011). It was also concluded that it could be possible for detonable mixtures of H_2 to build up in localized compartments of both types of dry containment and damage equipment. Therefore, it was decided that the potential effects of local H_2 burns should be evaluated on a plant-specific basis as a part of IPE.

3.8.3 Improved Plant Operations

Following the TMI-2 accident, the NRC shifted its emphasis from providing safety by relying on the traditional DBA to a multifaceted approach that also considered improved operations, human factors, realistic performance of systems, and PRA. The NRC program to improve plant operations consisted of many efforts, including regulatory actions to improve operational performance where it has fallen below expected standards, continued improvement of operational procedures, and expanding emergency operating procedures (EOPs) to include guidance on severe accident management strategies (NRC, 1988a).

3.8.4 Severe Accident Research

A severe accident research program (SARP) was an important part of the integration plan for closure of severe accident issues. The objectives of SARP were to identify and focus research necessary for sound regulatory decisions to be made within the framework of the integration plan and to prioritize the research activities needed to close severe accident issues. The overall near-term goals of the plan were to provide technical bases for assessing containment performance over the range of risk-significant accident sequences and to develop the capability to evaluate the efficacy of generic containment performance criteria. The long-term goals were to provide an improved understanding of severe accident phenomena and to develop improved methods for assessing fission product behavior and release during severe accidents.

As a part of SARP, analytical and experimental studies were performed to address many severe accident issues, including DCH, Mark I liner attack, and in-vessel steam explosion. A number of experiments were performed in support of DCH issue resolution for PWRs. Pilch et al. (1996) discuss the application of the Risk Oriented Accident Analysis Methodology (Theofanous, 1994) to address the DCH issue for 34 Westinghouse plants with large dry or subatmospheric containments.

Drywell liner melt-through (caused by direct contact with core debris) has been found to be the most important contributor to early containment failure for Mark I containments. This failure mode is only possible for Mark I containments, because the pedestal and drywell floor are at the same level and core debris can easily reach the containment liner. As a part of SARP, the NRC also sponsored analytical and experimental programs to address and resolve this “Mark I Liner Attack” issue. It was concluded that, in the presence of water, the probability of early containment failure by melt-attack of the liner is so low as to be considered physically unreasonable (Theofanous et al., 1989; Theofanous and Podowski, 1993).

Rapid steam pressure rise and missile resulting from in-vessel steam explosion had been identified as a potential challenge to the containment (alpha mode failure). However, a more recent (1996) assessment of this issue by an NRC-sponsored steam explosion review group (NRC, 1996a) concluded that alpha mode failure is of very low probability and that it is of little or no significance to the overall risk.

3.8.5 External Events

The NRC Severe Accident Policy Statement did not differentiate between events initiating within the power plant and events caused by external initiators, such as earthquakes, floods, and high winds. The evaluation of severe accidents initiated by external events proceeded in two phases. The first phase consisted of a Lawrence Livermore National Laboratory study to assess the margin that past design bases provided, relative to external events that were beyond the design basis and identify areas where an examination for external vulnerability might be needed (Kimura and Budnitz, 1987). The second phase consisted of developing specific guidance and criteria for each external hazard to be considered in the Individual Plant Examination of External Events (IPEEE).

On June 28, 1991, the NRC issued Supplement 4 to GL 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities." The supplement specifically requested licensees to perform an IPEEE to identify plant-specific severe accident vulnerabilities initiated by seismic events; internal fires; and high winds, floods, and other external initiating events, including accidents related to transportation, nearby facilities, and plant-unique hazards. The NRC received 70 IPEEE submittals covering all operating U.S. nuclear reactors. In addition to performing technical reviews of the IPEEE submittals, the NRC instituted a program to identify and document general perspectives and significant safety insights resulting from the IPEEE program. The NRC documented the results of this program in NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program," issued April 2002 (NRC, 2002).

3.8.6 Accident Management

The NRC recognized that the plant operating, and technical staff could take certain preparatory and recovery measures that could prevent or significantly mitigate the consequences of a severe accident (i.e., accident management). However, under the NRC program, licensees developed and implemented the accident management programs. The NRC worked with the industry to define the scope and attributes of a utility accident management plan and develop guidelines for plant-specific implementation. Nuclear Energy Institute (NEI) 91-04 (originally NUMARC 91-04), "Severe Accident Issue Closure Guidelines," issued December 1994 (NEI, 1994), contains binding implementing guidance relative to the formal industry position on severe accident management.

3.9 Severe Accident Mitigation Alternatives

In 1980, the NRC issued an interim policy statement on accident considerations under the National Environmental Policy Act of 1969 (NEPA), revising its policy to consider "the more severe kind of very low probability accidents that are physically possible in environmental impact assessments required by NEPA" (NRC, 1980f). The interim policy statement states that it is "the intent of the Commission that the staff takes steps to identify additional cases that might warrant early consideration of either additional features or other actions which would prevent or mitigate the consequences of serious accidents". These features have been referred to as severe accident mitigation design alternatives (SAMDAs) when applied at the design stage, or severe accident mitigation alternatives (SAMAs) when applied in the context of license renewal.

It was believed that the 1985 Severe Accident Policy Statement (discussed in section 3.7) was a sufficient basis for not requiring consideration of SAMDAs at the operating license review stage for previously constructed plants. However, a 1989 court decision (*Limerick Ecology Action v. NRC*, 869 F.d 719, 3rd Cir. 1989) ruled that such a policy statement was not sufficient to preclude consideration of SAMDAs and that such a consideration is required for plant operation.

It is understood that the regulatory programs and initiatives developed as a part of the Integration Plan for Closure of Severe Accident Issues (e.g., IPE, IPEEE) provide assurance that any major vulnerabilities to severe accidents have been identified and addressed and, therefore, no major plant modifications would be expected as a result of a SAMA analysis. As stated in NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," issued February 2023 (NRC, 1996b)—

> …the NRC expects that a site-specific consideration of severe accident mitigation for license renewal will only identify procedural and programmatic improvements (and perhaps minor hardware changes) as being cost-beneficial in reducing severe accident risk or consequence (NRC, 1996b).

3.10 Severe Accident Requirements for New Reactors

Severe accidents are addressed in 10 CFR 52.47, "Contents of applications; technical information," for standard design certifications through the requirement for (1) a design-specific PRA to be included in the application and (2) demonstration of compliance with any technically relevant portions of the TMI-related requirements set forth in 10 CFR 50.34(f). For LWR designs, 10 CFR 52.47(23) requires a description and analysis of design features for the prevention and mitigation of severe accidents (e.g., challenges to containment integrity caused by core concrete interaction, steam explosion, high pressure core-melt ejection, hydrogen combustion, and containment bypass). The NRC also augmented the hydrogen rule, 10 CFR 50.44, to include specific requirements for future reactor applicants and licensees. Such requirements include (1) ensuring a mixed atmosphere during significant beyond-design-basis accidents and (2) assuming hydrogen generation from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning in an analysis to demonstrate containment structural integrity.

In 1990, the staff prepared SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," (NRC 1990b), providing a list of issues and recommendations that would be fundamental to agency decisions on the acceptability of evolutionary advanced light-water reactors (ALWRs). The staff believed that the issues and recommendations were in keeping with the Commission's policy expectation that future designs for nuclear plants should achieve a higher standard of severe accident safety performance. In the staff requirements memorandum (SRM) dated June 26, 1990 (NRC, 1990c), the Commission approved, among other topics, the following staff recommendations as a basis for establishing regulatory guidance for evolutionary ALWR designs:

- *Core-concrete interaction:* Ability to cool core debris; approval of the general criteria that evolutionary ALWR designs (1) provide sufficient reactor-cavity floor space to enhance debris spreading, and (2) provide for quenching debris in the reactor cavity.
- *High-pressure core melt ejection:* ALWR designs should include a depressurization system and cavity design features to contain ejected core debris.
- *Containment performance:* Use a conditional containment failure probability of 0.1 or a deterministic containment performance goal that offers comparable protection in the evaluation of evolutionary ALWRs.

- *Advanced boiling-water reactor (ABWR) containment vent design:* Use a containment vent for ABWRs.

- *Equipment survivability:* Features provided only for severe accident protection need not be subject to the environmental qualification requirements in 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants"; quality assurance requirements in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"; and redundancy/diversity requirements in 10 CFR Part 50, Appendix A.

In the SRM dated January 28, 1992 (NRC, 1992a), the Commission approved the staff's recommendation to proceed with design-specific rulemakings through individual design certifications to resolve selected technical and severe accident issues for the GE ABWR and CE System 80+ designs.

In a letter dated May 17, 1991 (ACRS, 1991), the ACRS provided the Commission with proposed criteria to accommodate severe accidents in the containment designs of future LWRs. The Committee excluded the "evolutionary" LWRs for which designs were well advanced. However, the ACRS believed the new criteria could and should be adopted for use in the development and licensing of "passive" plant designs. The ACRS recommended that a set of new design requirements, including a definition of specific containment challenges posed by severe accidents, be promulgated through rulemaking into revisions and additions to 10 CFR Part 50, Appendix A. The staff agreed with the ACRS that ALWRs should consider severe accidents in their design. However, the staff did not agree with the ACRS approach of revising and amending the existing general design criteria (GDC) and believed that requirements addressing DBAs and severe accidents should be distinct and separate (NRC, 1992b). The staff agreed that severe accidents should be considered but not commensurate with the level of pedigree that DBAs demand.

3.11 Risk-Informed Regulations and Practices

In the early 1990s, the ACRS became concerned about the NRC's inconsistent use of PRA. In a letter dated July 19, 1991 (ACRS, 1992), on the consistent use of PRA, the ACRS acknowledged that "PRA can be a valuable tool for judging the quality of regulation, and for helping to ensure the optimal use of regulatory and industry resources." The Committee also stated that it "would have liked to see a deeper and more deliberate integration of the methodology into the NRC activities." The ACRS also pointed to issues such as the inconsistent use of conservatism and the lack of the treatment of uncertainties.

In response to the ACRS, the NRC chartered the PRA Working Group to address concerns identified by the ACRS with respect to the staff's uses of PRA. The working group issued its final report in 1993 (NRC, 1993) and identified the need for improvements in PRA guidance, training, methods, and data. In addition, the NRC chartered a regulatory review group to review processes, programs, and practices to identify the feasibility of substituting performance-based requirements and guidance founded on risk insights in place of prescriptive requirements (NRC, 2006a). These efforts led the Commission to issue a policy statement on the use of PRA (NRC, 1995a) so that the many potential applications of PRA could be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency.

In the 1995 policy statement on the use of PRA methods in NRC activities, the Commission promoted the increased use of probabilistic risk analysis "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 deterministic approach" (NRC, 1995a). The 1995 Commission policy statement stated the following:

> PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state of the art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule) (NRC, 1995a).

In issuing the policy statement, the Commission expected that its implementation would improve the regulatory process through (1) enhancement of decision-making using PRA insights, (2) more efficient use of agency resources, and (3) reduction in unnecessary burden on licensees. The possibility of removing unnecessary regulatory burden was a significant milestone in the evolution of regulations. Until the issuance of the policy statement, PRA insights had been used to add regulatory requirements. However, these insights also demonstrated that many of the conservative requirements of the "deterministic" approach did not contribute much to safety and, therefore, constituted an unnecessary regulatory burden. The possibility of removing such burden contributed to the wider acceptance of PRA by the licensees (Nourbakhsh, et al., 2018).

The 1995 PRA policy statement led the NRC to move toward a much-expanded use of PRAs in what is termed the risk-informed regulatory approach. Such an approach allowed PRA insights in concert with traditional "deterministic" analyses to be used for regulatory decision-making. The ACRS has been very supportive of the evolution toward a risk-informed and performance-based regulatory system, and has taken a leading role in considering some of the challenging issues that have arisen in this effort. In its letter dated May 19, 1999 (ACRS, 1999), on the role of defense in depth in a risk-informed regulatory system, the ACRS forwarded a paper (ACRS, 1999), prepared by several of its members and an ACRS Senior Fellow, which discussed two views ("structuralist" and "rationalist") of defense in depth, along with a preliminary proposal regarding the ACRS role in a risk-informed regulatory system. The ACRS motivation for this had arisen because of instances in which seemingly arbitrary appeals to defense in depth had been used to avoid making changes in regulations or regulatory practices that seemed appropriate in the light of results of quantitative risk analyses.

The NRC has applied information gained from PRAs extensively to complement other engineering analyses in improving issue-specific safety regulation and in changing the current licensing bases for individual plants. Using risk insights, the NRC has revised 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. Nourbakhsh, et al. (2018) includes a summary discussion on risk-informed regulations and practices.

4 POST-FUKUSHIMA SAFETY ENHANCEMENTS TO BETTER PREPARE AGAINST SEVERE ACCIDENTS

4.1 Lessons Learned from the Fukushima Dai-Ichi Events

On March 11, 2011, a magnitude 9.0 earthquake (Great Tōhoku Earthquake) struck Japan. It was soon followed by a tsunami, estimated to have exceeded 45 feet (14 meters) in height and resulting in extensive damage to the six nuclear power reactors at the Fukushima Dai-Ichi site.

In a tasking memorandum dated March 23, 2011 (NRC, 2011b), the NRC Chairman directed the staff to do the following:

> …establish a senior level agency task force to conduct a methodical and systematic review of our processes and regulations to determine whether the agency should make additional improvements to our regulatory system and make recommendations to the Commission for its policy direction (NRC, 2011b).

As a part of a long-term review, the task force was directed to "evaluate all technical and policy issues related to the event to identify potential research, generic issues, changes to the reactor oversight process, rulemakings, and adjustments to the regulatory framework that should be conducted by NRC" (NRC, 2011b).

The Near-Term Task Force (NTTF), established in response to the NRC Chairman's tasking memorandum, issued its report on July 12, 2011 (NRC, 2011c). The NTTF concluded that there was no imminent risk from continued operation and licensing activities. The NTTF also concluded that enhancements to safety and emergency preparedness (EP) were warranted and made the following recommendations for the Commission's consideration.

Recommendation 1: The NRC should establish a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations.

1.1 Draft a Commission policy statement that articulates a risk-informed defense-in-depth framework that includes extended design-basis requirements in the NRC's regulations as essential elements for ensuring adequate protection.

1.2 Initiate rulemaking to implement a risk-informed, defense-in-depth framework consistent with the above recommended Commission policy statement.

1.3 Modify the Regulatory Analysis Guidelines to implement the defense-in-depth philosophy more effectively in balance with the current emphasis on risk-based guidelines.

1.4 Evaluate the insights from the IPE and IPEEE efforts as summarized in NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," issued December 1997, and NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program," issued April 2002, to identify potential generic regulations or plant-specific regulatory requirements.

Recommendation 2: The NRC should require licensees to reevaluate and upgrade, as necessary, the design-basis seismic and flooding protection of structures, systems, and components (SSCs) for each operating reactor.

2.1 Order licensees to reevaluate the seismic and flooding hazards at their sites against current NRC requirements and guidance, and if necessary, update the design basis and SSCs important to safety to protect against the updated hazards.

2.2 Initiate rulemaking to require licensees to confirm seismic hazards and flooding hazards every 10 years and address any new and significant information. If necessary, update the design basis for SSCs important to safety to protect against the updated hazards.

2.3 Order licensees to perform seismic and flood protection walkdowns to identify and address plant-specific vulnerabilities and verify the adequacy of monitoring and maintenance for protection features such as watertight barriers and seals in the interim period until longer term actions are completed to update the design basis for external events.

Recommendation 3: As part of the longer term review, the NRC should evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.

Recommendation 4: The NRC should strengthen SBO mitigation capability at all operating and new reactors for design-basis and beyond-design-basis external events.

4.1 Initiate rulemaking to revise 10 CFR 50.63, "Loss of all alternating current power," to require each operating and new reactor licensee to (1) establish minimum coping time of 8 hours for a loss of all alternating current (ac) power, (2) establish the equipment, procedures, and training necessary to implement an "extended loss of all ac" coping time of 72 hours for core and spent fuel pool (SFP)_cooling and for reactor coolant system and primary containment integrity as needed, and (3) preplan and prestige offsite resources to support uninterrupted core and SFP cooling, and reactor coolant system and containment integrity as needed, including the ability to deliver the equipment to the site in the time period allowed for extended coping, under conditions involving significant degradation of offsite transportation infrastructure associated with significant natural disasters.

4.2 Order licensees to provide reasonable protection for equipment currently provided pursuant to 10 CFR 50.54(hh)(2) from the effects of design-basis external events and to add equipment as needed to address multiunit events while other requirements are being revised and implemented.

Recommendation 5: The NRC should require reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.

5.1 Order licensees to include a reliable hardened vent in BWR Mark I and Mark II containments.

5.2 Reevaluate the need for hardened vents for other containment designs, considering the insights from the Fukushima accident. Depending on the outcome of the reevaluation, appropriate regulatory action should be taken for any containment designs requiring hardened vents.

Recommendation 6: As part of the longer term review, the NRC should identify insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the Fukushima Dai-Ichi accident.

Recommendation 7: The NRC should enhance SFP makeup capability and instrumentation for the SFP.

7.1 Order licensees to provide sufficient safety-related instrumentation, able to withstand design-basis natural phenomena, to monitor key SFP parameters (i.e., water level, temperature, and area radiation levels) from the control room.

7.2 Order licensees to provide safety-related ac electrical power for the SFP makeup system.

7.3 Order licensees to revise their technical specifications to address requirements to have one train of onsite emergency electrical power operable for SFP makeup and SFP instrumentation when there is irradiated fuel in the SFP, regardless of the operational mode of the reactor.

7.4 Order licensees to have an installed seismically qualified means to spray water into the SFPs, including an easily accessible connection to supply the water (e.g., using a portable pump or pumper truck) at grade outside the building.

7.5 Initiate rulemaking or licensing activities or both to require the actions related to the SFP described in detailed recommendations 7.1–7.4.

Recommendation 8: The NRC should strengthen and integrate onsite emergency response capabilities such as EOPs, severe accident management guidelines (SAMGs), and extensive damage mitigation guidelines (EDMGs).

8.1 Order licensees to modify the EOP technical guidelines (required by Supplement 1, “Requirements for Emergency Response Capability,” to NUREG-0737, dated December 17, 1982 (GL 82-33)), to (1) include EOPs, SAMGs, and EDMGs in an integrated manner, (2) specify clear command and control strategies for their implementation, and (3) stipulate appropriate qualification and training for those who make decisions during emergencies.

8.2 Modify Section 5.0, “Administrative Controls,” of the Standard Technical Specifications for each operating reactor design to reference the approved EOP technical guidelines for that plant design.

8.3 Order licensees to modify each plant’s technical specifications to conform to the above changes.

8.4 Initiate rulemaking to require more realistic, hands-on training and exercises on SAMGs and EDMGs for all staff expected to implement the strategies and those licensee staff expected to make decisions during emergencies, including emergency coordinators and emergency directors.

Recommendation 9: The NRC should require that facility emergency plans address prolonged SBO and multiunit events.

9.1 Initiate rulemaking to require EP enhancements for multi-unit events in the following areas:

- personnel and staffing
- dose assessment capability
- training and exercises
- equipment and facilities

9.2 Initiate rulemaking to require EP enhancements for prolonged SBO in the following areas:

- communications capability
- Emergency Response Data System (ERDS) capability
- training and exercises
- equipment and facilities

9.3 Order licensees to do the following until rulemaking is complete:

- Determine and implement the required staff to fill all necessary positions for responding to a multiunit event.
- Add guidance to the emergency plan that documents how to perform a multiunit dose assessment (including releases from SFPs) using the licensee's site-specific dose assessment software and approach.
- Conduct periodic training and exercises for multiunit and prolonged SBO scenarios. Practice (simulate) the identification and acquisition of offsite resources, to the extent possible.
- Ensure that EP equipment and facilities are sufficient for dealing with multiunit and prolonged SBO scenarios.
- Provide a means to power communications equipment needed to communicate onsite (e.g., radios for response teams and between facilities) and offsite (e.g., cellular telephones, satellite telephones) during a prolonged SBO.
- Maintain ERDS capability throughout the accident.

9.4 Order licensees to complete the ERDS modernization initiative by June 2012 to ensure multiunit site monitoring capability.

Recommendation 10: As part of the longer term review, the NRC should pursue additional EP topics related to multiunit events and prolonged SBO.

10.1 Analyze current protective equipment requirements for emergency responders and guidance based upon insights from the accident at Fukushima.
10.2 Evaluate the command-and-control structure and the qualifications of decisionmakers to ensure that the proper level of authority and oversight exists in

the correct facility for a long-term SBO or multiunit accident or both.

- Concepts such as whether decision making authority is in the correct location (i.e., at the facility), whether currently licensed operators need to be integral to the emergency response organization outside of the control room (i.e., in the Technical Support Center), and whether licensee emergency directors should have a formal “license” qualification for severe accident management.

10.3 Evaluate ERDS to do the following:

- Determine an alternate method (e.g., via satellite) to transmit ERDS data that does not rely on hardwired infrastructure that could be unavailable during a severe natural disaster.

- Determine whether the data set currently being received from each site is sufficient for modern assessment needs.

- Determine whether ERDS should be required to transmit continuously so that no operator action is needed during an emergency.

Recommendation 11: As part of the longer term review, the NRC should pursue EP topics related to decision-making, radiation monitoring, and public education.

11.1 Study whether enhanced onsite emergency response resources are necessary to support the effective implementation of the licensees’ emergency plans, including the ability to deliver the equipment to the site under conditions involving significant natural events where degradation of offsite infrastructure or competing priorities for response resources could delay or prevent the arrival of offsite aid.

11.2 Work with FEMA, States, and other external stakeholders to evaluate insights from the implementation of EP at Fukushima to identify potential enhancements to the U.S. decision-making framework, including the concepts of recovery and reentry.

11.3 Study the efficacy of real-time radiation monitoring onsite and within the emergency planning zones (including consideration of ac independence and real-time availability on the Internet).

11.4 Conduct training, in coordination with the appropriate Federal partners, on radiation, radiation safety, and the appropriate use of potassium iodide in the local community around each nuclear power plant.

Recommendation 12: The NRC should strengthen regulatory oversight of licensee safety performance (i.e., the Reactor Oversight Process (ROP)) by focusing more attention on defense-in-depth requirements consistent with the recommended defense-in-depth framework.

12.1 Expand the scope of the annual ROP self-assessment and biennial ROP realignment to include defense-in-depth considerations more fully.

12.2 Enhance NRC staff training on severe accidents, including training resident inspectors on SAMGs.

4.2 Recommended Actions in Response to Fukushima Lessons Learned

In SRM-SECY-11-0093, “Staff Requirements—SECY-11-0093—Near-Term Report and Recommendations for Agency Actions Following the Events in Japan,” dated August 19, 2011

(NRC, 2011d), the Commission directed the staff "to engage promptly with stakeholders to review and assess the recommendations of the Near-Term Task Force in a comprehensive and holistic manner for the purpose of providing the Commission with fully-informed options and recommendations".

The Commission instructed the staff "to remain open to strategies and proposals presented by stakeholders, expert staff members, and others as it provides its recommendations to the Commission". The staff were also directed to do the following:

> …provide the Commission with a notation vote paper recommending a prioritization of the Task Force recommendations informed by the steering committee. This paper should reflect all regulatory actions to be taken by the staff to respond to Fukushima lessons learned, identify implementation challenges, include the technical and regulatory bases for the prioritization, identify any additional recommendations, and include a schedule and milestones with recommendations for appropriate stakeholder engagement and involvement of the ACRS. (NRC, 2011d).

The Commission also directed that the recommendation 1 "be pursued independent of any activities associated with the review of the other Task Force recommendations" (NRC, 2022d).

The SRM also directed the ACRS to "formally review all Task Force recommendations and the staff's evaluation and recommended prioritization of the Task Force recommendations and document its review in letter reports to the Commission" (NRC, 2011d).

As directed by the Commission, the staff reviewed the NTTF recommendations within the context of the existing regulatory framework and considered the various regulatory vehicles available to the NRC to implement the recommendations. First, the staff considered whether any of the NTTF findings identified an imminent hazard to public health and safety. As discussed in SECY-11-0124, "Recommended Actions to be Taken Without Delay from the Near-Term Task Force Report," dated September 9, 2011 (NRC, 2011e), the staff agreed with the NTTF that none of the findings rose to that level. Additionally, in SECY-11-0124, the staff identified a subset of the NTTF recommendations that should have been undertaken without unnecessary delay.

In its letter to the Commission on the initial ACRS review of NTTF report, dated October 13, 2011 (ACRS, 2011a), the Committee noted that "while complete understanding of the Fukushima Dai-Ichi accident will take many years, the NTTF Report and the staff's recommended actions to be taken without delay are appropriately focused on lessons learned from what is currently known." The Committee also believed that none of the recommendations enumerated would be negated, or rendered inappropriate, by the acquisition of new information. Hence, timely initiation of the staff's recommended actions to be taken without delay, along with corresponding additions or modifications included in the ACRS letter, was considered appropriate by the Committee (ACRS, 2011a).

In SECY-11-0137, "Prioritization of Recommended Actions to Be Taken In Response to Fukushima Lessons Learned," dated October 3, 2011 (NRC, 2011f), the staff proposed its prioritization of the Fukushima NTTF recommendations. As a result of the staff's prioritization and assessment process, the NTTF recommendations were prioritized into the three tiers.

Tier 1: The first tier consisted of NTTF recommendations that the staff determined should be started without unnecessary delay and for which sufficient resource flexibility exists, including the availability of critical skill sets. The Tier 1 recommendations were the following:

2.1 Seismic and flood hazard reevaluations

2.3 Seismic and flood walkdowns

4.1 SBO regulatory actions

4.2 Equipment covered under 10 CFR 50.54(hh)(2)

5.1 Reliable hardened vents for Mark I and Mark II containments

7.1 SFP instrumentation

8 Strengthening and integration of EOPs, SAMGs, and EDMGs

9.3 EP regulatory actions (staffing and communications)

Tier 2: The second tier consisted of NTTF recommendations that could not be initiated in the near term due to factors that included the need for further technical assessment and alignment, dependence on Tier 1 issues, or availability of critical skill sets. Those actions did not require long-term study and could be initiated when sufficient technical information and applicable resources become available. The Tier 2 recommendations were the following:

7 SFP makeup capability (7.2, 7.3, 7.4, and 7.5)

9.3 EP regulatory actions (the remaining portions of recommendation 9.3, except for ERDS capability addressed in Tier 3)

Tier 3. The third tier consists of NTTF recommendations that require further staff study to support a regulatory action, have an associated shorter term action that needs to be completed to inform the longer term action, are dependent on the availability of critical skill sets, or are dependent on the resolution of NTTF recommendation 1. The Tier 3 recommendations include all of the items identified for long-term evaluation in the NTTF report. In addition, the staff prioritized NTTF recommendations 2.2, 9.1, 9.2, 9.3 (ERDS capability), and 12 in Tier 3. The Tier 3 recommendations and associated prioritization logic are as follows:

2.2 Ten-year confirmation of seismic and flooding hazards (dependent on recommendation 2.1)

3 Potential enhancements to the capability to prevent or mitigate seismically induced fires and floods (long-term evaluation)

5.2 Reliable hardened vents for other containment designs (long-term evaluation)

6 Hydrogen control and mitigation inside containment or in other buildings (long-term evaluation)

9.1/9. 2 EP enhancements for prolonged SBO and multiunit events (dependent on availability of critical skill sets)

9.3 ERDS capability (related to long-term evaluation recommendation 10)

10 Additional EP topics for prolonged SBO and multiunit events (long-term evaluation)

11 EP topics for decision-making, radiation monitoring, and public education (long-term evaluation)

12.1 ROP modifications to reflect the recommended defense-in-depth framework (dependent on recommendation 1)

12.2 Staff training on severe accidents and resident inspector training on SAMGs (dependent on recommendation 8)

In SECY-11-0137, the staff also identified several additional issues with a clear nexus to the Fukushima Dai-Ichi event that may warrant regulatory action but were not included with the NTTF recommendations. Although the staff's assessment of these issues was incomplete at that time, several of those issues had already been judged to warrant further consideration and potential prioritization based on relative safety significance, nexus to NTTF recommendations, and other ongoing staff activities. The following additional recommendations warranted further consideration and potential prioritization:

- filtration of containment vents
- instrumentation for seismic monitoring
- basis of emergency planning zone size
- prestaging of potassium iodide beyond 10 miles
- transfer of spent fuel to dry cask storage
- loss of ultimate heat sink

In SRM-SECY-11-0124 dated October 18, 2011 (NRC, 2011g), the Commission approved the staff's proposed actions to implement without delay the NTTF recommendations as described in SECY-11-0124, subject to certain comments including the following:

- The process for implementing new or modified regulatory requirements or programs should be transparent and the regulatory mechanism (e.g., order, rulemaking, 10 CFR 50.54(f) letter, GL) used to impose them should be as clear and specific as possible when issued.
- As the staff evaluates Fukushima lessons learned and proposes modifications to the NRC's regulatory framework, the Commission encourages the staff to craft recommendations that continue to realize the strengths of a performance-based system as a guiding principle. In order to be effective, approaches should be flexible and able to accommodate a diverse range of circumstances and conditions. In consideration of events beyond the design basis, a regulatory approach founded on performance-based requirements will foster development of the most effective and efficient, site-specific mitigation strategies, similar to how the agency approached the approval of licensee response strategies for the "loss of large area" event under its B.5.b program.

In its letter to the Commission dated November 8, 2011 (ACRS, 2011b), the ACRS agreed with the staff's three-tier prioritization scheme. However, the Committee offered the following additional recommendations:

(1) Rulemaking activities related to strengthening of SBO mitigation capability should be expedited.

(2) Tier 1 recommendations should be expanded to include the additional immediate actions recommended in the October 13, 2011, ACRS report (ACRS 2011a) regarding flooding hazard reevaluations, integrated walkdowns, SBO, BWR hardened vents, shared ventilation systems, hydrogen control and mitigation, SFPs, and integration of onsite emergency actions (recommendations 1.a through 1.g, and 2.a through 2.d of October 13, 2011, ACRS report).

(3) NTTF recommendation 10.2 regarding evaluation of the command-and-control structure and qualifications of decision-makers should be initiated in parallel with Tier 1 activities related to integration of onsite emergency actions (NTTF recommendation 8).

(4) Tier 2 recommendations should be expanded to include the additional actions recommended in the October 13, 2011, ACRS report regarding enhancement of selected reactor and containment instrumentation, and the need to proactively engage in efforts to capture and analyze data from the Fukushima event (recommendations 2.e and 2.f of the October 13, 2011, ACRS report).

(5) Staff Tier 1 recommendation 7.1-2, “Develop and issue order to licensees to provide reliable SFP instrumentation,” should be reconsidered. Schedules for SFP instrumentation improvements and other modifications to the SFP should be informed by quantification of the contribution made by SFPs to overall plant risk.

In SRM-SECY-11-0137, dated December 15, 2011 (NRC, 2011h), the Commission approved the staff’s proposed prioritization of the NTTF recommendations and supported action on the Tier 1 and Tier 2 recommendations, subject to the direction contained in SRM-SECY-11-0124, and some additional comments including the following:

> In the absence of a fully developed justification for a proposed new requirement, the Commission finds it premature to initiate actions on the Near-Term Task Force recommendations under the premise of assuring or redefining the level of protection of public health and safety that should be required as adequate in accordance with the backfit rule. The Commission will evaluate the staff’s basis for imposing new requirements when documented in notation vote papers for any new requirements promulgated by orders or rulemaking.
>
> The staff should use INPO-11-005, “Special Report on the Nuclear Accident at the Fukushima Dai-Ichi Nuclear Power Station,” informed by country-specific considerations, as an input to its development of technical bases for any proposed regulatory changes.
>
> With respect to the six additional issues that the staff describes as having a clear nexus to the Fukushima Dai-Ichi event and that the staff’s indicates may warrant regulatory action but that were not included with the NTTF recommendations, the staff should provide the results of its determination of whether any regulatory action is recommended or necessary in the form of a SECY paper (information or notation vote, as appropriate). As with all other aspects of our Fukushima response, the Advisory Committee on Reactor Safeguards should provide its views of the staff’s approach. The paper should also address the November 8, 2011, ACRS Review of the Staff’s Prioritization, as appropriate.
>
> The staff should quickly shift the issue of “Filtration of Containment Vents” from the “additional issues” category and merge it with the Tier 1 issue of hardened vents for

Mark I and Mark II containments such that the analysis and interaction with stakeholders needed to inform a decision on whether filtered vents should be required can be performed concurrently with the development of the technical bases, acceptance criteria, and design expectations for reliable hardened vents (NRC, 2011h).

Consistent with Commission direction, the staff issued notation vote SECY-12-0025 on February 17, 2012 (NRC, 2012a), describing the proposed orders related to NTTF recommendations 4.2, 5.1, and 7.1, and the requests for information pertaining to NTTF recommendations 2.1, 2.3, and 9.3. SECY-12-0025 also describes the process developed by the staff to screen and disposition stakeholder recommendations and additional issues related to Fukushima beyond those included in the NTTF report.

4.3 Implementation of Lessons Learned from Fukushima

4.3.1 Mitigation Strategies Order, EA-12-049

On March 12, 2012, the NRC issued Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" (NRC, 2012b). The order addressed and expanded on recommendation 4.2 of the NTTF report.

The order requires a three-phased approach for mitigating beyond-design-basis external events to prevent core damage. The initial phase requires licensees to use installed equipment and resources to maintain or restore core cooling, containment, and SFP cooling. In the transition phase, licensees must provide sufficient portable onsite equipment and consumables to maintain or restore these functions until they can be maintained with offsite equipment and support. The final phase requires licensees to obtain sufficient offsite resources to sustain those functions indefinitely. Licensees must notify the NRC when they achieve full compliance with the order.

As of June 18, 2018, all operating power reactor units were in compliance with Order EA-12-049. Each plant installed new emergency response equipment, stored onsite and protected from natural hazards. NRC inspectors have verified that the strategies are in place at all U.S. nuclear power plants (NRC, 2021). The NRC accepted the U.S. nuclear power industry's proposed safety strategy—called diverse and flexible mitigation capability, or FLEX—which maintains long-term core and spent-fuel cooling and containment integrity with installed plant equipment that is protected from natural hazards, as well as backup portable onsite equipment. If necessary, similar equipment can be brought from off site. Additional equipment and resources are stored at two National Response Centers, ready to be deployed to a plant during an emergency (NRC, 2018).

4.3.2 Spent Fuel Pool Instrumentation Order, EA-12-051

On March 12, 2012, the NRC issued Order EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (NRC, 2012c), requiring all U.S. nuclear power plants to install reliable water-level measurement instrumentation in their SFPs to support prioritization of response activities between the reactor and the SFP in the event of an accident. The order addresses and expands on recommendation 7.1 of the NTTF report. The instrumentation must remotely monitor at least three distinct SFP water levels: (1) normal level, (2) low level but still high enough to shield workers above the pools from radiation, and (3) a very low level near the top of the spent fuel rods (indicating that more water should be added without delay).

As of June 30, 2017, all operating power reactor units are in compliance with Order EA-12-051. The NRC staff inspected for compliance with the SFP instrumentation order in conjunction with the inspections for Order EA-12-049 (NRC, 2021).

4.3.3 Reliable Hardened Containment Vents for Boiling-Water Reactors with Mark I & II Desings (Orders EA-12-050 and EA-13-109)

On March 12, 2012, the NRC issued Order EA-12-050, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents" (NRC, 2012d), requiring all licensees of operating BWRs with Mark I and II containments to install a reliable hardened vent. Subsequently, the staff considered the possibility of venting after reactor core damage occurs and provided its recommendation to the Commission in SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," dated November 26, 2012 (NRC, 2012e). In its report to the Commission on review of draft SECY-12-0157, the ACRS recommended the implementation of a performance-based approach, to reduce radioactive material releases as a needed defense-in-depth measure for BWR Mark I and Mark II containments. The ACRS further stated that severe accident capable vents are an essential part of any controlled venting strategy (ACRS, 2012).

In SRM-SECY-12-0157, dated March 19, 2013 (NRC, 2013a), the Commission directed the staff to require licensees with Mark I and II containments to "upgrade or replace the reliable hardened vents required by Order EA-12-050, with a containment venting system designed and installed to remain functional during severe accident conditions." The staff issued Order EA-13-109, "Issuance of Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," dated June 6, 2013 (NRC, 2013b), to ensure those vents would remain functional under the conditions that could exist in the event of reactor core damage. Order EA-13-109 superseded Order EA-12-050 and addressed and expanded on recommendation 5.1 of the NTTF report.

Order EA-13-109 contained two distinct phases of implementation. Phase 1 required affected licensees to upgrade the venting capabilities from the containment wetwell to provide a reliable hardened vent to help prevent core damage. The vent must also remain functional during severe accident conditions. Phase 2 required affected licensees to do one of the following:

- Increase protection for severe accident conditions through the installation of a reliable severe-accident-capable drywell vent system.
- Develop a reliable containment venting strategy that makes it unlikely that there would be the need to vent from the containment drywell during severe accident conditions.

As of June 21, 2019, all 17 applicable operating BWR sites subject to the order are in full compliance with the order EA-13-109 (NRC, 2021).

4.3.4 Flooding and Seismic Hazard Walkdowns

On March 12, 2012, the staff issued a request for information under 10 CFR 50.54(f), asking licensees, in part, to walk down their installed flooding protection, seismic protection, and flooding and seismic hazard mitigation features and review associated manual actions (NRC, 2012f). The request for information addressed and expanded on recommendation 2.3 of the NTTF report.

The licensees of operating reactors completed the plant walkdowns and submitted their walkdown reports. The staff assessments determined that the plant walkdowns consistently followed the intent of the NRC-endorsed guidance, thereby verifying that the walkdowns met the objectives of the 10 CFR 50.54(f) request for information (NRC, 2017).

4.3.5 Flooding Hazard Reevaluations

The NRC letter (NRC, 2012f) requesting information under 10 CFR 50.54(f) also asked licensees to use current regulatory guidance and methods to reevaluate the flooding hazards that could affect their sites.

In COMSECY-14-0037, "Integration of Mitigating Strategies for Beyond Design-Basis External Events and the Reevaluation of Flooding Hazards," issued November 21, 2014 (NRC, 2014), the staff requested Commission direction to define more clearly the relationship between Order EA-12-049, the related Mitigation of Beyond-Design-Basis Events (MBDBE) rulemaking, and the flooding hazard reevaluations. In SRM-COMSECY-14-0037 (NRC, 2015a), the Commission affirmed that licensees for operating nuclear power plants needed to address the reevaluated flooding hazards within their mitigating strategies. The Commission also directed the staff to provide a plan for achieving closure of the flooding portion of NTTF recommendation 2.1 to the Commission for its review and approval. The staff provided its plan the Commission in COMSECY-15-0019, "Closure Plan for the Reevaluation of Flooding Hazards for Operating Nuclear Power Plants," dated July 28, 2015 (NRC, 2015b). In SRM-COMSECY-15-0019, dated July 28, 2015 (NRC, 2015c), the Commission approved the plan.

The staff has been implementing the plan for ensuring that licensees address the reevaluated flooding hazards within mitigating strategies, as described in COMSECY-15-0019. On January 22, 2016, the NRC's Japan Lessons-Learned Division (JLD) issued interim staff guidance (ISG) JLD-ISG-2012-01, "Compliance with Order EA 12 049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design Basis External Events" (NRC, 2016a). This ISG endorses the NEI guidance document NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," issued December 2015 (NEI, 2015). Appendix G to this NEI document provides guidance for licensees to assess their mitigating strategies against the reevaluated flooding hazards.

As a result of flooding hazard evaluation, several nuclear power plant owners modified the protection of certain plant SSCs or identified alternative strategies to maintain the safety of the reactors in the event of a flooding event. Examples of flooding protection features include sandbags and/or inflatable berms, temporary and permanent pumps, site drainage (by manmade drains or natural grading), permanent flood walls, and watertight doors (similar to those found on ships) (NRC, 2018).

4.3.6 Seismic Hazard Reevaluations

Enclosure 1 of the 10 CFR 50.54(f) letter dated March 12, 2012 (NRC, 2012f), also requested that licensees reevaluate seismic hazards for their sites applying present-day methods and regulatory guidance used by the NRC staff when reviewing applications for early site permits and combined licenses.

The draft final MBDBE rule, provided in SECY-16-0142, "Draft Final Rule—Mitigation of Beyond Design Basis Events (RIN 3150 AJ49)," dated December 15, 2016 (NRC, 2016b), contained provisions that would have required mitigation strategies to address the reevaluated seismic hazard information on a generic basis. As echoed in the Affirmation Notice and SRM dated January 24, 2019 (NRC, 2019a), the Commission determined that addressing the reevaluated hazards in mitigation strategies on a generic basis was not needed for adequate protection but would instead be assessed on a plant-specific, case-by-case basis under the requirements of 10 CFR 50.109 and 10 CFR 52.98, "Finality of combined licenses; information requests".

As a result of seismic hazard evaluation, several nuclear power plant owners modified the protection of certain plant SSCs or identified alternative strategies to maintain the safety of the reactors in the event of a seismic event. Examples of seismic protection features include anchorages and restraints, spatial separation, and isolation systems and dampers (NRC, 2018).

4.4 The Risk Management Task Force

Shortly before the accident at the Fukushima Dai-Ichi nuclear power plants, George Apostolakis, then NRC Commissioner (also a former member and chairman of the ACRS), led the Risk Management Task Force 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 Risk Management Task Force report includes the following recommendation:

> …the NRC should establish through rulemaking a design-enhancement category of regulatory treatment for beyond-design-basis accidents. This category should use risk as a safety measure, be performance-based (including the provision for periodic updates), include consideration of costs, and be implemented on a site-specific basis (NRC, 2012g).

The Risk Management Task Force analysis was influenced by the events at Fukushima and the subsequent studies, including the NTTF and continuing discussions on the accident's implications for U.S. nuclear power plants (NRC, 2012g).

The Commission directed that the objectives of NTTF recommendation 1 be reevaluated, as appropriate, in the context of the Commission direction for a long-term Risk Management Regulatory Framework. The Commission later concluded that a proposed transition to a more fully integrated risk management regulatory framework should not be implemented at the time.

5 SEVERE ACCIDENT CONSIDERATIONS FOR ADVANCED REACTORS

5.1 Policy Statement on the Regulation of Advanced Reactors

On October 14, 2008, the Commission issued a policy statement on the regulation of advanced reactors (NRC, 2008). The Commission's 2008 policy statement reinforced and updated the policy statements on advanced reactors published in 1986 and 1994. Advanced reactors are considered to be those that are significantly different from the current generation of LWRs in operation or under construction, which include non-LWRs and other advanced reactor technologies. The 2008 policy statement identifies several attributes that could assist in establishing the acceptability or licensability of a proposed advanced reactor design, including the following attributes related directly to severe accidents:

- Designs that minimize the potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity, and independence in safety systems, with an emphasis on minimizing the potential for accidents over minimizing the consequences of such accidents.

- Designs that incorporate the defense-in-depth philosophy by maintaining multiple barriers against radiation release, and by reducing the potential for, and consequences of, severe accidents. (NRC, 2008).

5.2 Categorization of Event Sequences for Safety Evaluation Based on Probabilistic Risk Assessment

Many past efforts have sought to use PRA results in 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. Figure 1 compares the various approaches for the frequency-based categorization of event sequences. The sections below summarize these approaches.

5.2.1 Preapplication Safety Evaluation of Modular High-Temperature Gas-Cooled Reactor and Power Reactor Innovative Small Modular Reactor Designs

In the late 1980s and early 1990s, during preapplication safety evaluations of the Modular High-Temperature Gas-Cooled Reactor and Power Reactor Innovative Small Modern (PRISM) Liquid-Metal Reactor, it became necessary to consider a spectrum of accidents beyond the traditional LWR DBA envelope (NRC, 1989b; 1994). Therefore, a set of event categories was defined, corresponding to events that must be used for evaluating the safety characteristics of the proposed designs and assessing the adequacy of their containment systems and offsite emergency planning. Events in each category were selected deterministically, supplemented by insights gained from a PRA (NRC, 1989b).

Event Category I (EC-I) for advanced reactors was defined to be equivalent to the AOO class of events considered for LWRs. The frequency range for these events is approximately 10^{-2} per plant-year, or greater, which corresponds to the frequency of events that can be expected to occur one or more times during the life of the plant.

Event Category II (EC-II) was defined to be equivalent to the DBA category for LWRs. Consistent with the selection of an LWR DBA envelope, this category included events down to a frequency of 10^{-4} per plant-year. The value of 10^{-4}/year was based on ensuring that any event expected to occur over the lifetime of a population of reactors (100 reactors operating for 100 years) is included. However, the use of a lower value of 10^{-5} per plant-year was suggested to increase confidence that the collective risk of most potential DBAs is considered in the design and to account for uncertainties, particularly for a preapplication review.

Event Category III (EC-III) corresponded to severe events beyond the traditional DBA envelope to be used by designers in establishing the design bases for advanced reactors. The EC-III category included events down to an individual sequence frequency of approximately 10^{-7} per plant-year. The selection of 10^{-7} per plant-year was based on ensuring that the cumulative risk of several event sequences below 10^{-6} per plant-year are considered in assessing compliance with the Commission's proposed performance guideline of less than a 10^{-6} per plant-year frequency of a large release of radioactive material to the environment.

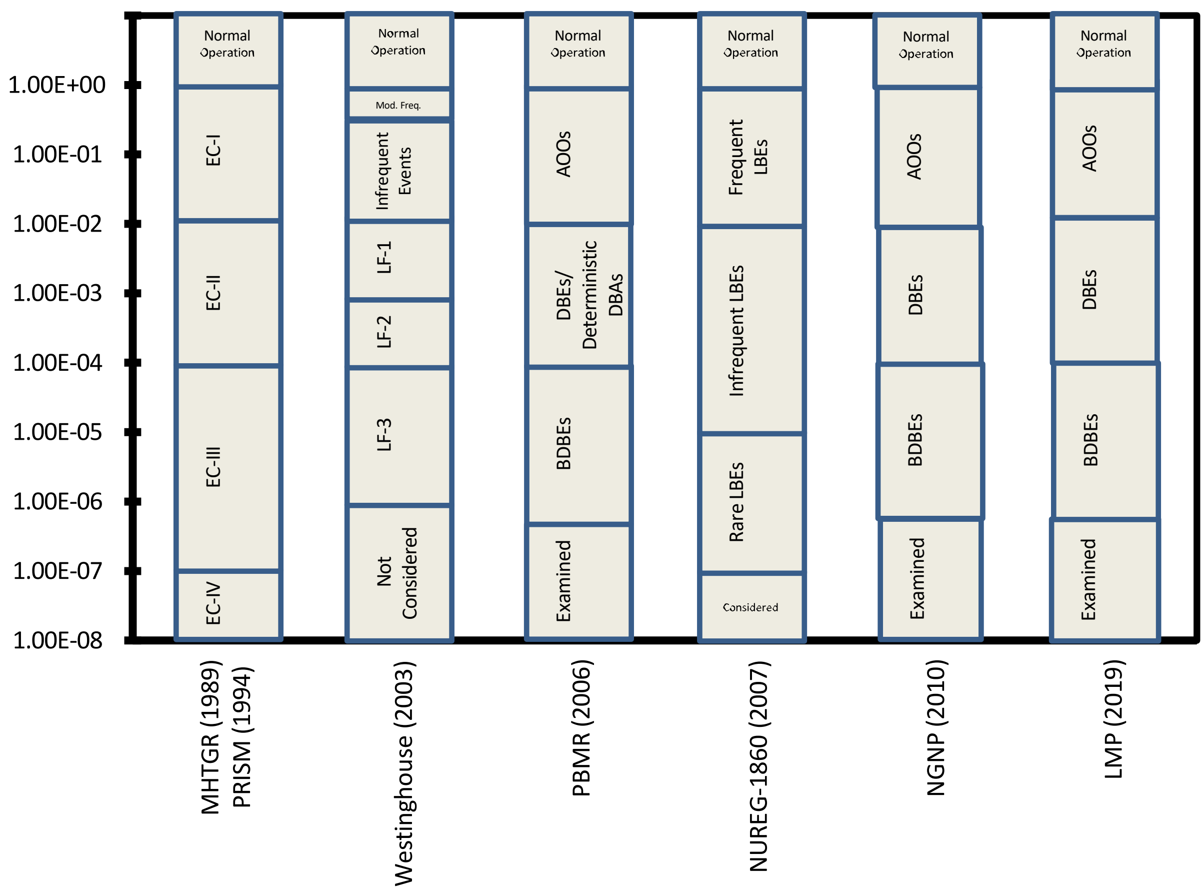

Figure 1 Comparison of Various Frequency-Based Categorizations of Event Sequences

Events in Event Category IV (EC-IV) were intended for use in assessment of the need for offsite emergency planning. EC-IV includes internal events of frequency similar to that of events considered in the basis for the emergency planning zones and requirements for LWRs (NRC, 1989b).

For evaluating the PRISM design, the Office of Nuclear Regulatory Research (66 RES) also developed a sequence categorization scheme, which relied on the type and number of systems, components, or operator failures to categorize sequences by qualitative risk, based on the likelihood of an initiating event. Subsequent failure probabilities were not needed to determine the sequence end state (or event category). RES also defined EC-IV to include scenarios of such low probabilities that detailed analysis would probably not be worthwhile. RES referred to these as “residual risk” scenarios (NRC, 1994). The RES event categorization scheme was not used for the PRISM preapplication evaluation.

5.2.2 Westinghouse Risk-Informed Safety Analysis Approach

In the traditional deterministic safety analyses the design-basis events (DBEs) are categorized by their initiating event frequencies. The use of coincident occurrences and single failures are also required in the analysis of the initiating event. In 2003, the Westinghouse Owners Group (WOG) proposed a risk-informed safety analysis approach for event frequency recategorization that considered events by their overall frequency, not just by that of the initiating event (Jacob and Rezendes, 2003). Correspondingly, the consequence acceptance criteria proposed for use in assessing conformance to regulatory criteria would be those associated with the overall event frequency, rather than the higher frequency of the initiating event alone.

The deterministic regulatory acceptance criteria are simply for two broad categories of events, namely, AOOs and accidents, whereas for the regulatory guidance documents (e.g., standard review plans), the evolution of the acceptance criteria has resulted in three broad categories of events: (1) moderate frequency events, (2) infrequent events, and (3) limiting faults. In the Westinghouse Risk-Informed Safety Analysis approach, the limiting faults category was broken down into three smaller frequency groups (Figure 1).

The NRC found the scope of Westinghouse report too broad for effective review and approval. Therefore, the WOG withdrew the report from staff review (WOG, 2003).

5.2.3 Licensing-Basis Event Selection for the Pebble Bed Modular Reactor

In 2006, PBMR (Pty) Ltd. proposed a structured, systematic, performance-based, and risk-informed process for selecting and analyzing licensing basis events (LBEs) for the Pebble Bed Modular Reactor (PBMR) (PMBR, 2006). The risk-informed licensing approach proposed for the PBMR included the definition of top-level regulatory criteria that provide frequency and dose limits for the LBEs. LBEs cover a spectrum of events from normal operation to rare, off-normal events. Each LBE is defined as a family of individual event sequences with a common initiating event, safety function response, and end state. The limits on the frequency ranges for the LBE categories were greater than 10^{-2} per plant year for AOOs, less than 10^{-2} and greater than 10^{-4} for DBEs, and less than 10^{-4} and greater than 5×10^{-7} for beyond-design-basis events (BDBEs). The events to 10^{-8} per plant-year are examined in the PRA to provide assurance that none are just below the minimum frequency of 5×10^{-7}. The LBEs in all three categories are evaluated individually to support the tasks of assessing the performance of SSCs with respect to safety functions in response to initiating events and collectively to demonstrate that the integrated risk of a multimodule plant design meets the NRC safety goals.

The deterministic DBAs are derived from the DBEs by assuming that only SSCs classified as safety related are available to mitigate the consequences. The public consequences of deterministic DBAs are based on mechanistic source terms and are conservatively calculated.

The upper bound consequence of each deterministic DBA must meet the consequence limit at the exclusion area boundary (EAB). BDBEs are also evaluated in developing emergency planning measures (PMBR, 2006).

Uncertainties of both frequency and consequence of each LBE are evaluated. The mean frequency is used to determine whether the event sequence family is an AOO, DBE, or BDBE. If the upper or lower bound (95th percentile or 5th percentile of the uncertainty distribution) on the LBE frequency straddles two or more regions, then the LBE is compared against the consequence criteria for each region. The mean, lower, and upper bound consequences are explicitly compared to the consequence criteria in all applicable LBE regions. The upper bound (95th percentile) for the DBE and deterministic DBA consequences must meet the 10 CFR 50.34, “Contents of applications; technical information,” dose limit at the EAB.

5.2.4 Feasibility for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing (NUREG-1860)

In mid-2000s, efforts were made to establish the feasibility of developing a risk-informed and performance-based regulatory structure for the licensing of future (advanced non-LWR) nuclear power plants. The results were documented in NUREG-1860, “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing,” issued December 2007 (NRC, 2007).

NUREG-1860 proposes a technology-neutral framework for selecting certain event sequences from the design-specific PRA for use in establishing plant design parameters for safe operation and equipment safety classification. These events, called LBEs, are sequences from the PRA that have to meet acceptance criteria related to the frequency-consequence (F-C) curve (figure 2) and additional deterministic criteria that depend on three broad ranges of mean accident frequency (NRC, 2007):

(1) frequent: > 10^{-2}/year

(2) infrequent: < 10^{-2}/year but > 10^{-5}/year)

(3) rare: < 10^{-5}/year but > 10^{-7}/year

LBEs are chosen by grouping similar accident sequences into an event class. Similar accident sequences are those that have similar initiating events and display similar accident behavior in terms of system failures and/or phenomena and lead to similar source terms (NRC, 2007). The LBE representing a group of sequences is assigned the 95th percentile frequency of the most likely sequence in the group and the 95th percentile consequence of the most challenging sequence in the group.

In its report on development of a technology-neutral regulatory framework, dated September 26, 2007 (ACRS, 2007), 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 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.” Additionally, the Committee was 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 ACRS also recommended that "the framework should recognize accident prevention as a fundamental regulatory goal and should specify a quantitative limit on the frequency of an accident." The Committee noted that "in technology neutral terms, an accident can be defined as the release of radionuclides within the plant significantly in excess of normal operating limits".

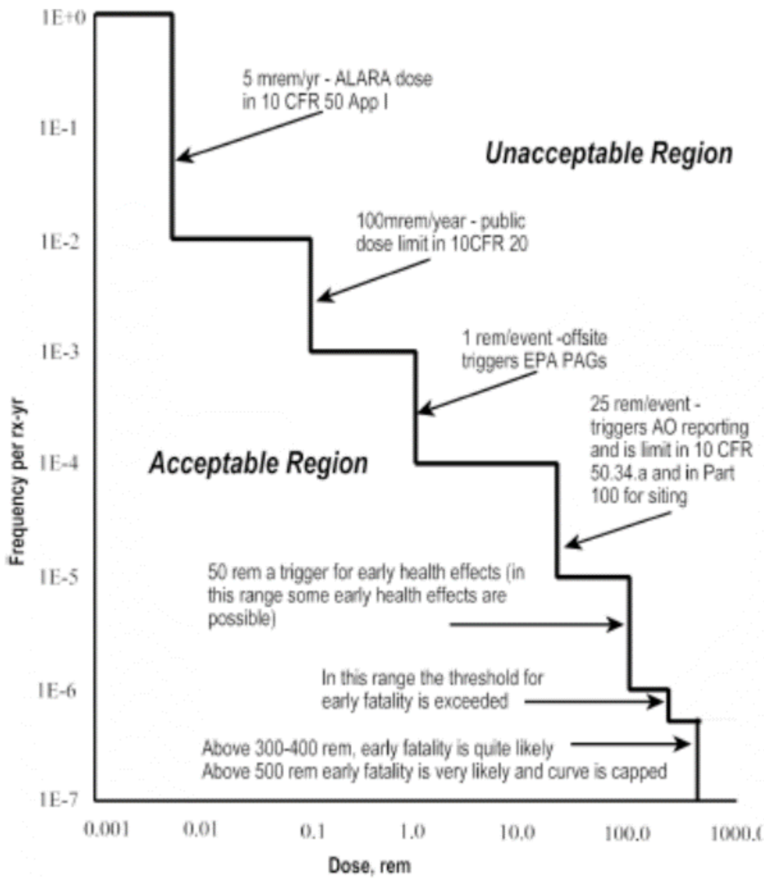

Figure 2 The F-C Limit Curve from NUREG-1860 (NRC, 2007)

In its April 30, 2008, report on draft NUREG-1902, "Next Generation Nuclear Plant Licensing Strategy Report," the ACRS noted that "the DOE [U.S. Department of Energy] and NRC should take this opportunity to exercise the risk-informed and performance-based technology-neutral framework for developing licensing requirements for any future commercial development of NGNP [Next Generation Nuclear Plant] design" (ACRS, 2008).

The technology-neutral framework proposes that the event sequences that make up the LBEs are selected from a complete PRA of the plant design covering both internal and external events and all modes of operation. However, Johnson and Apostolakis (2012) concluded that the goals sets forth in NUREG-1860 are extremely stringent with regard to seismic risk and other external events and cannot be met for a traditional LWR.

5.2.5 Next Generation Nuclear Plant Licensing Basis Event Selection White Paper

In a 2010 white paper (INL, 2010), Idaho National Laboratory (INL) outlined a systematic performance-based and risk-informed methodology for selecting and classifying LBEs for the NGNP design. The proposed methodology for categorizing and selecting LBEs for NGNP licensing is somewhat similar to the one previously proposed for PBMR (discussed in section 5.3).

The limits on the frequency range for the LBE categories (AOOs, DBEs, and BDBEs) proposed for the NGNP are the same as those proposed for the PBMR.

The DBAs are derived from the DBEs by assuming that only SSCs classified as safety related are available to mitigate the consequences (INL, 2010).

The mean frequencies are used to determine the event categories. If the upper or lower bound on the LBE frequency overlaps two or more regions, then the LBE is compared against the consequence criteria for each region. The mean, lower, and upper bound consequences are explicitly compared to the consequence criteria in all applicable LBE regions. The upper bound for the DBE and DBA consequences must meet the dose limit at the EAB (INL, 2010).

5.2.6 Licensing Modernization Project Approach to Selecting Licensing Basis Events

The Licensing Modernization Project (LMP)—led by Southern Company, coordinated by the NEI, and cost-shared by the DOE—has proposed changes to specific elements of the current licensing framework and a process for implementing the proposals. The LMP's objective has been to assist the NRC in developing regulatory guidance for licensing advanced non-LWR plants (Southern Company, 2019). Section 5.3 contains more discussion of the LMP.

To derive the appropriate list of LBEs, the LMP uses a set of F-C criteria, referred to as the F-C Target (Figure 3).

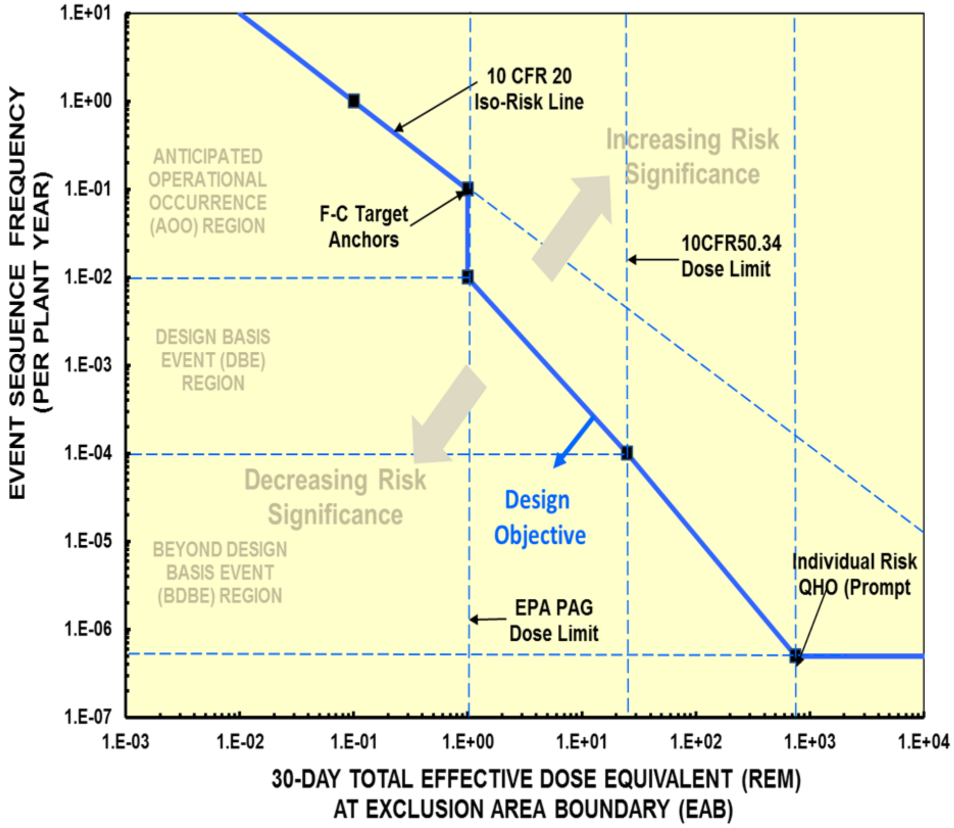

Figure 3 F-C Target Proposed for the LMP (Southern Company, 2019)

The F-C Target provides a general reference to assess events, SSC's, and programmatic controls in terms of sensitivities and available margins. LBE categories are based on mean event sequence frequency of occurence per plant-year. AOOs are off-normal events that are expected to occur with frequencies exceeding 10^{-2}/plant-year, where a plant may comprise multiple reactor modules (NEI, 2019). DBEs are less ^{2}/plant-year. BDBEs are events with frequencies less than 10^{-4}/plant-year but with upper bound frequencies greater than $5^{\times}10^{-7}$/plant-year. LBEs may or may not involve the release of radioactive material and may involve two or more reactor modules or radionuclide sources (NEI, 2019).

The F-C Target for high-frequency AOOs down to a frequency of 10^{-1}/plant-year are based on an iso-risk profile defined by the annual exposure limits in 10 CFR Part 20, "Standards for Protection Against Radiation", (100millirem (mrem)/plant-year). The F-C Target for lower frequency AOOs at frequencies of 10^{-1}/plant-year down to $10^{-2/}$plant-year are set at a reference value of 1 rem, corresponding with EPA Protective Action Guide (PAG) limits and consistent with the goal of avoiding the need for offsite emergency response for any AOO. The F-C Target for DBEs range from 1 rem at 10^{-2}/plant-year to 25 rem 10^{-4}/plant-year, with the dose calculated at the EAB for the 30-day period following the onset of the release. This aligns the lowest frequency DEBs to the limits in 10 CFR 50.34 and provides continuity to the lower end of the AOO criteria. The F-C Target for the BDBEs ranges from 25 rem at 10^{-4}/plant year to 750 rem at 5×10^{-7}/plant-year to ensure the quantitative health objective (QHO) for early health effects is not exceeded for individual BDBEs (NEI, 2019).

5.3 Industry-Led Licensing Modernization Project

The LMP has proposed changes to specific elements of the current licensing framework and a process for implementation of the proposals. The interactions between the NRC staff and the LMP resulted in the submittal of the NEI 18-04, Revision 1, "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," issued August 2019 (NEI, 2019), focused on identifying LBEs, categorizing and establishing performance criteria for SSCs, and evaluating defense in depth for advanced reactor designs.

In a notation vote paper SECY-19-0117, "Technology-Inclusive, Risk Informed, and Performance Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," dated December 17, 2019 (NRC, 2019b), the staff discussed potential policy issues associated with the LMP methodology and recommended that the Commission find that the use of the methodology described in NEI 18-04 is a reasonable approach for establishing key parts of the licensing basis for non-LWRs. In SRM-SECY-19-0117, dated May 26, 2020 (NRC, 2020a), the Commission found that using the methodology is a reasonable approach to support the licensing of non-LWRs.

In June 2020, the NRC issued Regulatory Guide 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors" (NRC, 2020b). The regulatory guide endorses, with clarifications, the methodology documented in NEI 18-04.

The ACRS reviewed draft SECY-19-0117, the associated draft regulatory guide (DG-1353), and NEI 18-04. In its report dated March 19, 2019 (ACRS, 2019), the Committee concluded that "the approach has matured to the point of being ready for application" and stated that "the guidance proposed in DG-1353 is adequate to support implementation of the approach described in the SECY paper, with the exception that guidance for developing mechanistic source terms should be expanded".

5.4 Rulemaking on a Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors

As required by the Nuclear Energy Innovation and Modernization Act (NEIMA) (NEIMA, 2019), the NRC has begun rulemaking efforts to establish a technology-inclusive regulatory framework for optional use by commercial advanced nuclear reactor applicants for new reactor license applications. This rulemaking would create 10 CFR Part 53, "Licensing and Regulation of Advanced Nuclear Reactors." NEIMA defines the term "advanced nuclear reactor" as "a nuclear fission or fusion reactor, including a prototype plant (as defined in sections 50.2 and 52.1 of title 10, *Code of Federal Regulations* (as in effect on the date of enactment of this Act)), with significant improvements compared to commercial nuclear reactors under construction as of the date of enactment of this Act, including improvements such as….".

The staff presented its proposed plan for this rulemaking to the Commission for approval in SECY-20-0032, “Rulemaking Plan on “Risk Informed, Technology Inclusive Regulatory Framework for Advanced Reactors,” dated April 13, 2020 (NRC, 2020c). In SRM-SECY-20-0032, dated October 2, 2020 (NRC, 2020d), the Commission approved the staff’s proposed approach and directed the staff to publish the final rule by October 2024.

The regulatory requirements developed in the 10 CFR Part 53 rulemaking would use methods of evaluation, including risk-informed and performance-based methods, that are flexible and practicable for application to a variety of advanced reactor technologies.

Although the proposed 10 CFR Part 53 is expected to support licensing under either the construction permit and operating license processes described in 10 CFR Part 50 or the licensing, certification, and approval processes described in 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” the staff intended to build 10 CFR Part 53 with as few connections as possible to any prescriptive or programmatic criteria specified in 10 CFR Part 50 and 10 CFR Part 52. However, the staff intended to incorporate attributes of 10 CFR Part 50 and 10 CFR Part 52 that the staff and stakeholders determine would facilitate regulatory reliability and clarity due to demonstrated efficacy in prior regulatory activities (NRC, 2020c).

The staff has focused the rulemaking on risk-informed functional requirements, building on existing NRC requirements, Commission policy statements, and recent activities undertaken to implement the NRC’s vision and strategy for non-LWRs (NRC, 2020c). The staff is also building on ongoing activities, such as those described in SECY-19-0117, which in turn describes the methodology of the LMP.

In its October 21, 2020, report on 10 CFR Part 53 licensing and regulation of advanced nuclear reactors (ACRS, 2020), the ACRS supported the staff plan for developing the new 10 CFR Part 53 rule and alerted the staff to several issues that the Committee intends to follow closely as the rulemaking process evolves, including the following:

> Novel aspects of new technologies make the identification of hazards, initiating events, and scenarios challenging; systematic searches will be needed. A process for gaining confidence in safety calculations will also be needed that focuses on the theoretical and experimental basis for fully understanding the associated physics and chemistry of possible scenarios. The levels of design and knowledgebase completeness affect our ability to have confidence in the conservatism of assumptions in traditional transient and accident analyses as well as the calculated margins. Likewise, the lack of completeness provides a challenge for probabilistic risk assessment (PRA), which should assess the resulting uncertainties explicitly.
>
> To address uncertainties caused by limited information, there is no substitute for critical examination of the design, its safety behavior, and all aspects of operations, starting from a blank sheet of paper to avoid bias. This implies a need for compensatory measures such as alternative systematic searches for hazards, initiating events, and accident scenarios with no preconceptions that could limit the creative process.
>
> Incorporating the concept of 10 CFR Part 50 GDC into the framework of the proposed 10 CFR Part 53 rulemaking is important. The GDC was developed

and included in 10 CFR Part 50 to improve the predictability and efficiency of NRC reviews of licensing applications. The GDC established requirements for design, fabrication, construction, testing, and performance, to ensure that needed structures, systems, and components (SSCs) remain functional during and following identified design basis events.

We look forward to engaging the staff on criteria similar to the GDC of 10 CFR Part 50, Appendix A. We know the staff is considering alternatives and we have already commented on the advanced reactor design criteria (ARDC) developed as part of the Non-LWR Vision and Strategy program. It is our expectation that the staff will find a logic structure that makes clear the links among critical safety functions, functional groups of GDC, and the detailed GDC themselves, and will be able to apply it to high level design criteria expected to be codified in 10 CFR Part 53 and associated detailed guidance (ACRS, 2020).

On March 1, 2023, the NRC staff issued SECY-23-0021, “Proposed Rule: Risk Informed, Technology Inclusive Regulatory Framework for Advanced Reactors (RIN 3150 AK31)” (NRC, 2023), to obtain Commission approval to publish a draft proposed rule in the Federal Register that would amend regulations to establish a voluntary risk-informed, performance-based, and technology-inclusive regulatory framework for commercial nuclear plants.

In SRM-SECY-23-0021, dated March 4, 2024 (NRC, 2024), the Commission approved, with certain exceptions and clarifications, the draft proposed rule that would amend the CFR to establish a voluntary risk-informed, performance-based, and technology-inclusive regulatory framework for commercial nuclear plants as proposed by the staff. The Commission disapproved the inclusion of the safety objectives in draft proposed rule.

6 SUMMARY AND CONCLUSIONS

This paper has discussed historical perspectives and insights on severe accident regulatory decisions. It also presented an overview of the past observations and recommendations by the ACRS, regarding the protection against severe accidents.

The potential consequences of severe reactor accidents have been the subject of interest and study since the earliest days of reactor development. However, there was a general agreement that the probability of occurrence of severe accidents in nuclear power reactors was exceedingly low. Severe accident regulatory decisions mostly dealt with reducing the likelihood of such a serious accident rather than coping with one. This approach assumed 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.

The 1979 accident at TMI-2 led to the reexamination of the design basis and the consideration of regulations for protection against severe accidents.

The 1985 Commission policy statement on severe reactor accidents (NRC, 1985) regarding future designs and existing plants affirmed the Commission's belief that a new design for a nuclear power plant can be shown to be acceptable for severe accident concerns if it meets certain criteria and procedural requirements, including "completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that may add to the assurance of no undue risk to public health and safety".

The 1995 PRA policy statement led the NRC to move toward a much-expanded use of PRAs, in what is termed a risk-informed regulatory approach. Such an approach allowed PRA insights in concert with traditional, "deterministic" analyses to be used for regulatory decision-making.

The 2008 Commission policy statement on the regulation of advanced reactors (NRC, 2008) reinforced and updated the previous policy statements on advanced reactors published in 1986 and 1994. The 2008 policy statement identified several attributes that could assist in establishing the acceptability or licensability of a proposed advanced reactor design, including attributes that minimize the potential for severe accidents and their consequences.

On March 11, 2011, a 9.0-magnitude earthquake, Great Tōhoku, struck Japan and was soon followed by a tsunami estimated to have exceeded 45 feet (14 meters) in height, resulting in extensive damage to the six nuclear power reactors at the Fukushima Dai-Ichi site. Following this accident, the NRC required significant enhancements to U.S. commercial nuclear power plants, including (1) adding capabilities to maintain key plant safety functions following a large-scale natural disaster, (2) updating evaluations on the potential impact from seismic and flooding events, (3) adding new equipment to better handle potential reactor core damage events, and (4) strengthening EP capabilities.

As required by NEIMA, the NRC has begun rulemaking efforts to establish a technology-inclusive regulatory framework for optional use by commercial advanced nuclear reactor applicants for new reactor license applications.

This rulemaking would create 10 CFR Part 53, whose regulatory requirements would use methods of evaluation, including risk-informed and performance-based methods, that are flexible and practicable for application to a variety of advanced reactor technologies.

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11. ABSTRACT (200 words or less)

As part of its commitment to the U.S. Nuclear Regulatory Commission's knowledge management efforts, the Advisory Committee on Reactor Safeguards (ACRS) has begun an initiative to capture the Committee's institutional knowledge and memory. An important motivation for this initiative is to increase the effectiveness and efficiency of the Committee's review process by providing ready access to background information, insights, and understanding related to technical and regulatory issues. This report provides historical perspectives and insights on severe accident regulatory decisions. It also presents an overview of prior ACRS observations and recommendations regarding protection against severe accidents.

The views expressed in this report are solely those of the author and do not necessarily represent the views of the ACRS.

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