*To be presented at the International Meeting on Severe Accident Assessment and Management: Lessons Learned from Fukushima Dai-ichi, San Diego, CA, November 11-15, 2012*

# Insights and Perspectives on Severe Accident Regulatory Decisions

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*The events at the Fukushima Dai-ichi Nuclear Power Station in Japan have provided an impetus for the reexamination of regulations for protection against severe accidents. This paper begins with historical perspectives and insights on severe accident regulatory decisions. An overview of the past ACRS observations and recommendations regarding the protection against severe accidents is also presented.*

## I. 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 to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) and categorized in Regulatory Guide (RG) 1.70, are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit. Plants should be able to handle the full range of these AOOs with no fuel damage, and be returned to operation.

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

A severe accident is a very low frequency event, brought about by multiple failures, which 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. For severe accidents, historically, only a few direct regulatory requirements such as emergency planning were instituted. Severe accident regulatory decisions have mostly dealt with reducing the likelihood of such a serious accident rather than coping with one. This approach was based on the assumption that, because of the "defense in depth" design philosophy, such accidents are of sufficiently low probability that mitigation of their consequences is not necessary for public safety. The 1979 accident at Three Mile Island Unit 2 (TMI-2) led to the re-examination of the design basis and the 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) issued by U.S. 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 inert the boiling water reactor (BWR) Mark-I and Mark-II containments and install igniters for hydrogen control in BWR Mark-III and ice-condenser containments.

The events at the Fukushima Dai-ichi Nuclear Power Station in Japan have provided an impetus for the re-examination of regulations for protection against severe accidents. All technical and policy issues related to the event are being evaluated to identify potential rulemakings and adjustments to the regulatory framework. This paper begins with historical perspectives and insights on severe accident regulatory decisions. An overview of the past ACRS observations and recommendations regarding the protection against severe accidents is also presented.

## II. EARLY YEARS OF OPERATION OF NUCLEAR POWER PLANTS

The potential consequences of severe reactor accidents have been the subject of interest and 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) [1], "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants."

Beginning in 1961, the AEC began defining a standard regulatory prescription to licensing. The 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. Part 100 was developed, in part, based on the assumptions that an upper limit of fission product release could be estimated and the

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

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 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. 10 CFR Part 100 requires that the suitability of the reactor site be judged based in part on a postulated fission product release (into the containment) associated with a "substantial meltdown" of the core. For currently licensed nuclear power plants, the characteristics of the fission product release from the core into the containment are set forth in Regulatory Guides 1.3 and 1.4 and have been derived from the 1962 AEC report (TID-14844) [2], "Calculation of Distance Factors for Power and Test Reactor Sites." The TID-14844 assumed a core meltdown and instantaneous release of all noble gases, fifty percent of the iodine, and one percent of the other core particulate materials (solids) to the containment atmosphere.

The use of TID release assumptions has not been confined to a determination of site suitability alone. The regulatory applications of releases of this type cover a wide range, including the basis for (1) performance requirements of important fission-product clean-up 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 proposed plants increased significantly in size, 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. At the "prodding" of ACRS, the AEC established a special task force to look into the problem of core meltdown in 1966 [3]. The task force, chaired by William K. Ergen, a former ACRS member, issued its report in October 1967[4]. The report offered assurances about the improbability of a core meltdown and the reliability of emergency core cooling system (ECCS) designs, but it also acknowledged that a loss-of-coolant accident (LOCA) could cause a breach of containment if the ECCS 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" [3].

In an ACRS letter on the task force 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." The ACRS also stated that the proposal in the task force report for study of preventive measures to be made effective prior to loss of containment integrity to minimize the ultimate hazard is 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." The task force 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) [5] and the beginning of the science of probabilistic risk assessment as applied to nuclear power plant safety [6].

## III. LESSONS LEARNED FROM THE TMI-2 ACCIDENT

The March 28, 1979 accident at Three Mile Island Unit 2 (TMI-2), led to the 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 loss-of-coolant accident (LOCA), whose consequences should have been bounded by those of a large-break LOCA, but became much more severe due to misunderstanding of the event by the operators. 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 the late John Kemeny, then president of Dartmouth College, to investigate what had happened and its possible impact on the health and safety of the public and plant personnel. The President's Commission on the Accident at TMI issued its report in October 1979 [7], which contained several recommendations. Among them:

> "*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.*"

In May 1979, the Nuclear Regulatory Commission 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 [8] in particular states that:

> *"…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."*

The Lessons Learned Task Force led to the publication of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" [9] and NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report" [10]. In its December 13, 1979 letter on the final TMI-2 Lessons Learned Task Force Final Report, the ACRS gave general support to many of the Task Force recommendations. With regard to design features for core-damage and core-melt accidents, the Committee believed that "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." The ACRS also made some comments and recommendations on several matters not directly addressed in NUREG-0578 or NUREG-0585. Among them:

> *"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 nonseismic Class 1 equipment, and the suitability of emergency procedures for earthquakes."*

After the TMI-2 accident the NRC decided that power reactor licensing should not continue until the assessment of the accident had been substantially completed and comprehensive improvements in both the operation and regulation of nuclear power plants had been set in motion [11]. About nine months after the accident, the NRC proposed a post TMI-2 action plan for utilities. In developing the action plan, the various recommendations and possible actions were assessed and either rejected, adopted or modified. On June 16, 1980, the NRC issued its policy statement regarding the requirements to be met for current operating license applications [11]. The requirements were derived from the NRC's Action Plan (NUREG-0660) [12] and were documented in NUREG-0694, "TMI-Related Requirements for New Operating Licenses" [13].

In its January 15, 1980 letter on Draft NUREG-0660, the ACRS believed the Plan was comprehensive, but not selective. The Committee stated that "this comprehensiveness serves to dilute the items important to safety, and therefore important to termination of the licensing pause." The Committee further stated that "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." Only specific items from NUREG-0660 were approved by the Commission for implementation at reactors. Those specific items, including additional information about schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions, were documented in NUREG-0737, "Clarification of TMI Action Plan Requirements" [14].

As a result of the degraded core accident at TMI-2 and subsequent reevaluation of regulatory processes, NRC published an advance notice of proposed rulemaking on October 2, 1980 [15], announcing that it was considering amending its regulations to determine to what extent commercial nuclear power plants should be designed to cope with reactor accidents beyond those considered in the current "design basis accident" approach. In particular, 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 the 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 (IDCOR) program to provide an industry perspective for any rulemaking activities that might proceed. The Notice of Proposed Rulemaking was later withdrawn (see Section IX).

## IV. HYDROGEN RULE

The first significant regulatory action for severe accident mitigation was the hydrogen rule (10 CFR 50.44) issued by U.S. NRC 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 boiling water reactor (BWR) Mark-I and Mark-II containments and install igniters for hydrogen control in BWR Mark-III and the ice condenser containments. The pressurized water reactor (PWR) plants with large dry containments (including those operating with a sub-atmospheric internal pressure) were exempted from hydrogen control, because of the large volume of their containments.

## V. 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) states 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 (LPZ) surrounding a nuclear power plant site. In 1970, explicit requirements for plans to cope with emergencies were published in 10 CFR 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."

In an April 8, 1975 ACRS letter on emergency planning, 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 that "*sound emergency planning requires the ability to cope with a wide range of accident situations*." The Committee further stated that "*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*." The Committee recommended that "*such scenarios need to be developed and drills incorporating appropriate responses should be conducted*."

In 1976, an ad hoc Task Force of the Conference of (State) Radiation Control Program Directors passed a resolution requesting NRC to "make a determination of the most severe accident basis for which radiological emergency response plans should be developed by offsite agencies" [16]. In November 1976, a Task Force consisting of NRC and EPA representatives was convened to address this Conference request and related issues. The recommendations of the Task Force on Emergency Planning were published as NUREG-0396 [16]. 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 upon knowledge of the timing, fission product release characteristics and potential consequences of a spectrum of accidents. A number of accident descriptions were considered, including the severe accident release categories of the Reactor Safety Study.

Although the TMI 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 Kemeny Commission, 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 emergency planning rule (10 CFR 50.47) 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.

## VI. 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 [17], "Technical Bases for Estimating Fission Product Behavior during LWR Accidents." The development of this report was prompted, in part, by the December 21, 1980 letter, 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 Reactor Safety Study assumptions including the conclusion that cesium iodide (CsI) would be the expected predominant iodine chemical form under most postulated LWR accident conditions.

The potential impact of the NUREG-0772 findings on reactor regulation was also examined and the results were documented in NUREG-0771 [18]. These studies formed the basis for the designation of five accident groups as being 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. Light Water Reactors (LWRs) was published in NUREG-0956 [19]. This reassessment involved reviewing experimental and analytical results from severe accident research programs sponsored by the NRC and the nuclear industry.

The NUREG-1150 study [20] 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 the inclusion of the 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 design basis accidents at nuclear reactors. In 1995, the NRC published NUREG-1465 [21], which defined an alternative accident source term for regulatory applications. The release fractions for the alternative accident source terms were derived from the insights and simplifications of the NUREG-1150 source term analyses documented in NUREG/CR-5747 [22].

## VII. SAFETY GOAL POLICY STATEMENT

In 1979, The ACRS recommended that consideration be given by the NRC to the establishment of quantitative safety goals for nuclear power reactors. In its May 16, 1979 letter on quantitative safety goals, the ACRS recognized the difficulties and uncertainties in the quantification of 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 ACRS was at the forefront of the development of quantitative safety goals. The first set of trial goals (NUREG-0739) [23] was developed by the ACRS in 1980. These safety goals were the basis for the later NRC work on the development of an NRC Safety Goal Policy in 1983 [24].

The NRC safety goal policy statement focuses on the 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 which 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. "*

## VIII. BACKFIT RULE

The backfit rule, 10 CFR 50.109, was first adopted by the AEC in 1970. 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 costs of the backfit could not be considered under the Atomic Energy Act, versus other backfits which represented an enhancement to safety beyond what may 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.

## IX. SEVERE ACCIDENT POLICY STATEMENT

In the 1985 Commission policy statement on severe reactor accidents regarding future designs and existing plants [25], the Commission concluded, based on available information, that existing plants posed no undue risk to the 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 which might 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 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."

## X. INTEGRATION PLAN FOR CLOSURE OF SEVERE ACCIDENT ISSUES

Subsequent to issuance of the Severe Accident Policy Statement, a number of separate programs on severe accidents were being pursued by the NRC. So in 1988, the NRC coordinated these programs with an "Integration Plan for Closure of Severe Accident Issues" [26]. That plan consisted of six main elements: (1) Individual Plant Examinations; (2) Containment Performance Improvements; (3) Improved Plant Operations; (4) Severe Accident Research; (5) External Events; and (6) Accident Management.

### X.A. Individual Plant Examinations

As a key part of the implementation of the severe accident policy statement, the NRC issued Generic Letter 88-20 in 1988 [27], requesting that each licensee conduct an individual plant examination (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 then examined the IPE submittals to determine what the collective IPE results imply 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" [28].

### X.B. 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 NRC then concluded that there are generic severe accident challenges to each LWR containment type that should be assessed to determine whether additional regulatory guidance or requirements concerning needed containment features was 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 [26].

At the conclusion of the MARK I Containment Performance Improvement Program, a number of plant modifications that could substantially enhance the plants' capability to both prevent and mitigate the consequences

of severe accidents were identified [29]. The recommended improvements included (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 the recommended safety improvements, with one exception, that was, hardened wetwell vent capability, be evaluated by licensees as part of the IPE Program. Also, the NRC issued Generic Letter 89-16 indicating that it would approve hardened vents for licensees who propose to install them and perform a backfit analysis for licensees who do not propose to install them.

MARK II containment vulnerabilities and potential improvements were similar to those identified for MARK I containments [29]. 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 (EPGs) entitled "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 considers MARK I improvements, other than the hardened vent, as part of its IPE.

For Mark III plants, potential improvements were also similar to those for MARK I plants [29]. However due to the relatively large volume of the MARK III containment, the need for venting was found to be less likely than for the MARK I containment. Also, some MARK III plants already had the capability to vent through a hardened system. A potential vulnerability for MARK III plants involved station blackout, during which the igniters would be inoperable. Under these conditions, a detonable mixture of H2 could develop which could be ignited upon restoration of power. A potential improvement considered for MARK III containments was a backup power supply in order to be able to use igniters during a station blackout. 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 [29]. The results of risk analysis for Sequoyah (a PWR with 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 direct containment heating (DCH) phenomena. Risk assessments varied considerably in their characterizations of DCH contribution to containment failure. As part of the Accident Management Program (see Section X.F), full or partial depressurization of the RCS 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 are operating to control the large amount of H2 that may be produced. Containment failure resulting from uncontrolled H2 burns or detonations was found to be a potentially important failure mode for ice condenser containments [29]. This could occur in station blackout events if power to the H2 igniter system is lost, high concentrations of H2 are produced as a result of core degradation, and power is then restored at a later time. The NRC requested each licensee with ice condenser containment consider, as part of its IPE, the insights and improvements identified in the containment performance improvements program.

The NRC did not identify any generic improvements that would have been applicable to all dry containments. H2 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 sub-atmospheric containments [29]. It was also concluded that it could be possible for detonable mixtures of H2 to build-up in localized compartments of both types of dry containments and damage equipment. Therefore, it was decided that the potential effects of local H2 burns should be evaluated on a plant-specific basis as a part of IPE.

### X.C. Improved Plant Operations

Following the TMI-2 accident the NRC shifted its emphasis from providing safety by relying on the traditional design basis approach to a multifaceted approach which also considered improved operations, human factors considerations, realistic performance of systems, and probabilistic risk assessment. 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 EOPs to include guidance on severe accident management strategies [29].

### X.D. Severe Accident Research

A severe accident research program (SARP) was an integral part of the integration plan for closure of severe accident issues. The objective of the SARP was 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 the 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 of the plan 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 direct containment heating (DCH), Mark I liner attack, and in-vessel steam explosion. A number of experiments were performed in support of DCH issue resolution for PWRs. Reference [30] provides discussions on application of Risk Oriented Accident Analysis Methodology (ROAAM) [31] to address the DCH issue for 34 Westinghouse plants with large dry or sub-atmospheric 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 so-called "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 [32].

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 assessment of this issue in 1996 by an NRC sponsored steam explosion review group [33] concluded that alpha mode failure is of very low probability and that it is of little or no significance to the overall risk.

### X.E. 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 (LLNL) 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 [34]. The second phase consisted of developing specific guidance and criteria for each external hazard to be considered in the Individual Plant Examination for External Events.

In 1991, the NRC issued Supplement 4 to GL 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities." That 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 results of this program were documented in NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program" [35].

### X.F. Accident Management

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

## XI. SEVERE ACCIDENT MITIGATION ALTERNATIVES

In 1980, NRC issued an interim policy statement on accident considerations under the National Environmental Policy Act (NEPA) of 1969, revising its policy for "considering the more severe kind of very low probability accidents that are physically possible in environmental impact assessments required by NEPA" [36]. The interim policy statement states that it is "*the intent of the Commission that the staff take 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 SAMAs when applied in the context of license renewal.

It was believed that the 1985 Severe Accident Policy Statement (see Section IX) was a sufficient basis for not requiring a 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 a 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 [37], "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."

## XII. SEVERE ACCIDENT REQUIREMENTS FOR NEW REACTORS

Severe accidents are addressed in 10 CFR 52.47 for standard design certifications through the requirement for a design-specific PRA to be included in the application and 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, providing a list of issues and recommendations which 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 achieve a higher standard of severe accident safety performance. In a staff requirement memorandum (SRM) dated June 26, 1990; 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 (CCFP) of 0.1 or a deterministic containment performance goal that offers comparable protection in the evaluation of evolutionary ALWRs.

- ABWR Containment Vent Design; Use a containment vent for advanced boiling water reactors (ABWRs).

- Equipment Survivability; Features provided only for severe accident protection need not be subject to the 10 CFR 50.49 envirormental qua1ificattion requirements, 10 CFR Part 50, Appendix B, quality assurance requirements, and 10 CFR Part 50, Appendix A, redundancy/ diversity requirements.

In the January 28, 1992 SRM, 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 May 17, 1991 letter, 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 the "passive" plant designs. The ACRS recommended that a set of new design requirements, that would include a definition of specific containment challenges posed by severe accidents, be promulgated through rulemaking into revisions and additions to Appendix A, "General Design Criteria" (GDC), to 10 CFR Part 50. The staff agreed with the ACRS in that advanced LWRs should consider severe accidents in their design. However, the staff did not agree with the ACRS approach of revising and amending the existing GDC and believed that requirements addressing design basis accidents and severe accidents should be distinct and separate [38]. The staff agreed that severe accidents should be considered, but not commensurate with the level of pedigree that design basis accidents demand [38].

## XIII. RISK-INFORMED REGULATIONS AND PRACTICES

The U.S. NRC led the development of quantitative risk analysis for nuclear power plants. Though PRAs had been used extensively in the past, they were usually limited to a variety of applications on a case by case basis as deemed necessary or useful. The NRC started moving toward a much expanded use of PRAs in what is termed risk-informed regulatory approach. In 1995, the NRC adopted a policy that promotes increasing the use of probabilistic risk analysis in all regulatory matters to the extent supported by the state-of-the-art to complement the deterministic approach. The current regulatory framework is based largely, but not entirely on a deterministic approach that employs safety margins, operating experience, accident analyses, and a defense-in-depth philosophy. Probabilistic assessment of the risk is used to support and inform regulatory decisions on generic and plant-specific issues.

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 modified its oversight process and its requirements for maintenance (10 CFR 50.65). The NRC has made other revisions to its reactor regulations (10 CFR Part 50) to focus requirements on programs and activities that are most risk significant. However, these revisions provide alternatives that are strictly voluntary to current requirements.

The ACRS has been very supportive of the evolution toward a risk-informed and performance-based regulatory system**.** ACRS has taken a leading role in considering some of the challenging issues that have arisen in this effort. In its May 19, 1999 letter on the Role of Defense in Depth in a Risk-Informed Regulatory System, ACRS forwarded a paper, prepared by several of its members and an ACRS Senior Fellow, in which two ("Structuralist" and "Rationalist") views of defense-in-depth were discussed along with a preliminary proposal regarding its 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 ACRS actively participated in development of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The publication of RG 1.174 in 1998 was a major milestone in the NRC initiative to risk-inform the regulations. RG 1.174 introduced the concept of an integrated decision-making process that had as inputs risk information, considerations of defense in depth, and sufficient safety margins.

In early 2011, at the request of the NRC Chairman, Commissioner George Apostolakis (a former member and Chairman of the ACRS) led a 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 findings and recommendations of the task force (named the Risk Management Task Force (RMTF)) have been documented in a report entitled "A proposed Risk Management Regulatory Framework" [39]. This report includes the recommendation that "*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*."

The accident at the Fukushima Dai-ichi nuclear power plants in Japan occurred shortly after the RMTF was established. As noted in the RMTF report [39], "*the team's analysis has been influenced by the events at Fukushima and the subsequent studies, including NRC Near-Term Task Force (see Section XIV), and continuing discussions on the accident's implications for U.S. nuclear power plants*."

## XIV. LESSONS LEARNED FROM THE FUKUSHIMA DAI-ICHI EVENTS

On March 11, 2011, a 9.0-magnitude earthquake 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.

In a March 23, 2011 tasking memorandum, the NRC Chairman directed the staff "to 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." 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." The Near-Term Task Force (NTTF), established in response to the NRC Chairman's tasking memorandum, issued its report [40] on July 12, 2011. 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 are warranted and made several recommendations for Commission consideration.

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. The NTTF recommendations which the staff determined should be started without unnecessary delay included:

- Seismic and flood hazard reevaluations,
- Station blackout (SBO) regulatory actions,
- Reliable hardened vents for Mark I and Mark II containments, and
- Strengthening and integration of emergency operating procedures, severe accident management guidelines (SAMGs), and extensive damage mitigation guidelines

In its October 13, 2011 letter to the Commission on initial ACRS review of NTTF report, 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.*"

## XV. CONCLUSIONS

Historical perspectives and insights on severe accident regulatory decisions were provided. An overview of the past ACRS observations and recommendations regarding the protection against severe accidents was also presented. The events at the Fukushima Daiichi Nuclear Power Station in Japan have provided an impetus for the re-examination of regulations for protection against severe accidents. The U.S. NRC efforts on evaluating the technical and policy issues related to the event to identify potential rulemakings and adjustments to the regulatory framework were also discussed.

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