July 30, 1997 SECY-97-171

FOR: The Commissioners

FROM: L. Joseph Callan /s/
Executive Director for Operations

SUBJECT: CONSIDERATION OF SEVERE ACCIDENT RISK IN NRC REGULATORY DECISIONS

PURPOSE:

To provide the Commission with background on how severe accident risk has been considered by the Commission in making past regulatory decisions and how the risk from severe accidents is being considered for potential future actions.

BACKGROUND:

In a February 12, 1997, staff requirements memorandum (SRM), the Commission directed the staff to provide a summary paper that details consideration of severe accident risk, both in past regulatory decisions or rules and potential future actions. The Commission also directed the staff to provide an assessment, and recommendations if appropriate, for formalizing the agency's position on consideration of severe accident risk.

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DISCUSSION:

The Commission has been considering severe accidents (accidents more severe than design basis accidents in which substantial damage is done to the reactor core whether or not there are serious offsite consequences) in its regulatory decisions and actions since its early days. These include decisions in which severe accidents have been considered directly in making regulatory decisions (i.e., specific regulatory requirements to address accidents more severe than design basis accidents) and decisions in which severe accidents have been considered more indirectly in making decisions (e.g., by considering the results of cost/benefit analyses in the decision-making process). The probability of a severe accident occurring, as well as the potential consequences of the accident, were considered qualitatively by agency decision makers during the early regulatory decisions. These qualitative considerations involved the use of engineering judgement and were made in the context of a deterministic consideration of accidents beyond the design basis. The "risk" of severe accidents, as that term is generally used in current NRC lexicon as the quantitative product of a probability times a consequence, was not utilized by the agency until relatively recently.

Prior to the accident at Three Mile Island Unit 2 (TMI), the focus of the Commission was on design basis accidents. Following the accident at TMI, there was a shift in the Commission's focus to provide greater consideration of the risks from severe accidents in its decision making. Furthermore, in the past, although the Commission has considered accidents beyond the design basis in its decision making and in establishing requirements, licensees and applicants have not generally been required to consider them explicitly in the design of their facilities. More recently, the Commission has required licensees and applicants to evaluate their plants for severe accident vulnerabilities. The following discussion provides examples of the Commission's consideration of accidents more severe than design basis accidents.

In its early consideration of severe accidents, the Commission recognized that accidents beyond the design basis, although low in probability, could occur. Typically, reliance was placed upon the concept of "defense-in-depth" to minimize the likelihood and consequences of such accidents. For example, consideration of accidents beyond the design basis was clearly a consideration in the Commission's decisions on reactor siting criteria. In the original issuance of Part 100, the Statement of Considerations noted that:

> "Further, since accidents of greater potential hazard than those commonly postulated as representing an upper limit are conceivable, although highly improbable, it was considered desirable to provide

for
protection against excessive exposure doses to people in large centers, where effective protective measures might not be feasible ... Hence, the population center distance was added as a site requirement...."

In addition, the source term used for assessment of the Part 100 dose guidelines was based upon a "substantial meltdown of the core."

In 1971, a rulemaking was proposed to implement a scheme for classifying reactor accidents into 9 classes with the level of severity increasing from Class 1 (trivial incidents) to Class 9 (severe accidents involving core meltdown). It was intended that accidents in Classes 1-8 be used by applicants in the preparation of environmental reports for nuclear reactor power plants, and these accidents were to be included in the Atomic Energy Commission's review. Applicants were not to be required to analyze Class 9 accidents due to their perceived low likelihood of occurrence. In light of the TMI accident in 1979 and a number of criticisms directed at the approach described in the proposed rulemaking announcement, the rulemaking was formally withdrawn in June of 1980.

In an early study, in 1957, the AEC published WASH-740, "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants." This study evaluated the potential consequences for several accident scenarios and discussed in broad terms a range of likelihoods for the occurrence of such accidents. In 1975, the completion of WASH-1400, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," represented a significant advance in the use of Probabilistic Risk Assessment (PRA) methods. This first quantitative perspective of the likelihood of severe accidents replaced the previous qualitative perspective of "highly improbable" considered in the decisions on siting. The risk insights provided by WASH-1400 were also considered in the staff's development of recommendations on establishment of emergency planning in NUREG-0396 (December 1978) which served as the basis for emergency planning requirements.
However, as a result of criticisms arising from the WASH-1400 Risk Assessment Review Group, the Commission issued a policy statement in January 1979, addressing the issues raised by the Review Group. These issues included concerns over calculational methods, data base quality and the importance

of
uncertainties in the interpretation of the PRA results generated by the study.
The January 1979 policy statement accepted the Review Group Report's conclusion that absolute values of the risks presented in WASH-1400 should not be used uncritically either in the regulatory process or for public policy purposes. The Commission did not regard the Reactor Safety Study's numerical estimate of the overall risk of reactor accidents as reliable. However, the Commission stated "Taking due account of the reservations expressed in the Review Group Report and in its presentation to the Commission, the Commission supports the extended use of probabilistic risk assessment in regulatory decisionmaking." Further, the Commission provided additional detailed instructions to the NRC staff concerning continued use of risk assessment techniques and results in response to specific criticisms raised by the Risk Assessment Review Group. The accident at TMI in March of 1979 elevated the consideration of severe accidents in the Commission's decision-making process.

Following the accident at TMI, a number of actions were taken to specifically address severe accidents. TMI Action Plan items were documented in NUREG-0660,"NRC Action Plan Developed As a Result of the TMI-2 Accident" (May 1980) and NUREG-0737, "Clarification of TMI Action Plan Requirements" (January 1983). This plan included tasks to address core degradation beyond design basis conditions with the aim of reducing individual and societal risk. Some items in the plan involved the performance of specialized reviews of the operating plants' designs and operations. Other items in the plan, such as core melt behavior (including the subsequent fuel-coolant interactions and core-concrete interactions) and the effects of potential hydrogen combustion on containment integrity, were incorporated into NRC's Severe Accident Research Program. TMI Action Plan Item II.B.6 resulted in risk studies of operating reactors in areas of high population density (i.e., Zion, Limerick, Indian Point) to determine what additional measures or design changes could be implemented that could further reduce the probability or the consequences of a severe accident. Action Plan Item II.B.8 resulted in the Commission issuing a revision to 10 CFR 50.44 that added requirements for light-water cooled power reactors to include the capability to control hydrogen gas following a postulated loss-of-coolant accident.

Also as a result of TMI, 10 CFR 50.34(f), "Additional TMI-related requirements," was promulgated in 1982 for a limited set of plants listed in the rule whose designs were currently under review by the staff. Part 50.34(f)(1)(i) of this rule section required that a plant/site specific probabilistic risk assessment be performed to seek plant design improvements - in effect for protection against severe accidents. This was the first incidence where the Commission required applicants to evaluate their designs for severe accidents using PRA. While none of the specific plants listed in the new rule ever completed construction and licensing, Part 50.34(f)(1)(i) was applied to the General Electric GESSAR-II, the Combustion Engineering System 80, and the Westinghouse RESAR-SP/90 designs during the Commission's review of these standard reactor designs. Numerous potential design improvements aimed at reducing the risk from severe accidents were evaluated for the GESSAR-II design, the first application of the new rule. It was also stipulated in Part 50.34(f)(1)(i) that the improvements must be significant, practical and not impact significantly on the plant, resulting in the inclusion of cost/benefit in consideration of potential design enhancements.
In a later rulemaking, the Commission issued Part 52 to implement the standard design certification process that includes by reference the requirement that all future reactor design applications include a PRA thereby providing an evaluation of the design for severe accidents. In Part 52.47(ii), under "Contents of applications," it is stated that certified design applications must include a design-specific probabilistic risk assessment in addition to a demonstration of compliance with any technically relevant portions of 10 CFR 50.34(f).

Following TMI, increased use of risk methodology was made in selected regulatory programs. One of these early applications was in the use of risk insights as part of the NRC's Systematic Evaluation Program (SEP). This activity involved assessing some of the earliest nuclear power plants against current regulatory requirements. PRA was utilized as one element in assessing the risk and safety significance of deviations from current requirements and providing insights on risk effectiveness of proposed modifications.

While the Commission performed evaluations of various potential design improvements to ensure that cost effective means for reducing plant risk from severe accidents were considered, these evaluations were performed only for safety interests and were not considered in the context of minimizing the impact on the environment for purposes of the National Environmental Policy Act of 1969. Then, a decision by the U.S. Court of Appeals in 1989 resulted in a review of Severe Accident Management Design Alternatives (SAMDAs) as a part of the environmental impact review for the Limerick plant operating license. A similar evaluation was subsequently performed for Commanche Peak Units 1 and 2 and for Watts Bar Unit 1. Since the Limerick decision, all initial operating license proceedings and design certification rulemakings have considered SAMDAs as part of the staff's safety review in order to support compliance with NEPA. However, the need for SAMDAs is addressed in the Environmental Impact Statement for the plant rather than in the Safety Evaluation Report (SER).

As the Commission continued to evaluate potential new requirements for plants to deal with accidents that were considered to be beyond the normal design basis, it issued two rules in the 1980s that dealt with Anticipated Transients Without Scram (ATWS, July 1984) and Station Blackout (June 1988) each of which had been identified in previous safety studies as potentially being an important contributor to risk. In promulgating these rules, the Commission considered the reduction in risk to the public associated with the implementation of the rule and the costs to implement the new requirements.
In both cases, the Commission established deterministic requirements that, when met, served to reduce the risk from severe accidents.

In August 1985, the Commission published its "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants." In the policy statement, the Commission said that it had concluded that existing plants pose no undue risk to public health and safety and saw no basis for immediate action on generic rulemaking or other regulatory actions to deal with severe accidents. However, the Commission indicated its intention to initiate a systematic examination of each nuclear power plant for possible significant risk contributors. In the policy statement, the Commission also said that it fully expected that designers of new plants would achieve a higher standard of severe accident performance than prior designs.

At the same time that the Commission was developing its Severe Accident Policy Statement that addressed the procedures that the Commission intended to use to resolve severe accident issues, the Commission was also developing its Safety Goal Policy to establish goals that broadly defined an acceptable level of risk. In 1986, the Commission issued its "Policy Statement on Safety Goals for the Operations of Nuclear Power Plants." This policy statement focused on risks to the public from the release of radioactive materials to the environment for normal operations as well as from accidents. The Commission established two qualitative safety goals which are supported by two quantitative objectives. The quantitative objectives are to be used to determine that the safety goals have been achieved.

Prior to TMI, the staff began to explore the use of PRA methods and cost benefit information to prioritize generic safety issues. After TMI, the staff began to systematically rely on regulatory analyses for NRC regulatory decisions. Regulatory analyses, by their nature, evaluate proposed actions that may be needed to protect the public health and safety, and as such, consideration of severe accident risks has consistently been an integral part of these analyses. The principal element of a regulatory analysis is an evaluation of the costs and benefits, in which health and safety benefits are estimated based on PRA information on the change in risk. In this evaluation, the benefit of averting the consequences of severe accidents (averted person-rem) is converted to dollars based on NRC's policy of using a $2000 per person-rem conversion factor. This factor allows a direct comparison between the potential health and safety benefits and the costs of a proposed regulatory initiative. In addition, part of the regulatory analysis includes a safety goal evaluation which could eliminate a proposed requirement from further consideration if the predicted reduction in risk resulting from implementation of the requirement is below a threshold screening value.

These safety goal evaluations rely on PRA results in which the estimated change in the core damage frequency per reactor-year and the conditional probability of early containment failure and bypass are compared to safety goal screening criteria. The screening criteria developed by the staff were derived from a subsidiary safety goal for CDF of 1x10-4 per reactor-year and a conditional probability of early containment failure or bypass (CPCFB) of 0.1 consistent with the Commission's guidance to the staff for evaluating the Evolutionary Light Water Reactor designs. This process that evolved for evaluating costs and benefits of potential plant improvements was to become a major consideration in the implementation of the Backfit Rule, 10 CFR 50.109. Details regarding the preparation of regulatory analyses are provided in NUREG/BR-0058, Revision 2, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission."

In November 1988, the staff, with Commission approval, issued Generic Letter 88-20 that asked licensees to conduct Individual Plant Evaluations (IPEs) to look for plant-specific vulnerabilities to severe accidents. The staff issued a supplement to Generic Letter 88-20 in June 1991 that asked licensees to evaluate their plants for vulnerabilities to severe accidents from external events (IPEEEs, e.g., fires, seismic events). Although the Commission initially left it to licensees to identify improvements to their plants, the Commission reserved the right to impose additional requirements using cost/benefit criteria under the Backfit Rule. The staff has provided the Commission with the status of and progress on IPE and IPEEE reviews on a regular basis. The staff has essentially completed its review of the IPEs submitted by the licensees. In total, the licensees have reported that approximately 500 improvements in plant design or operation have been implemented as a result of the IPE effort. The IPEEE submittals are presently undergoing review.

In 1990, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," was published. Internally initiated accidents and externally initiated accidents (two plants) up through offsite consequences and quantitative risk were evaluated. This document reflected the state-of-the-art understanding of severe accident phenomenology and analysis

methods
including uncertainties in plant risk.  This assessment and its documentation
have provided a model for subsequent PRA studies including those used in the
design certification reviews for the ABWR, System 80+ and AP600 ALWR designs.

As indicated above, Part 52 mandates that a PRA accompany any future plant
application for design certification.  In addition to the severe accident
evaluation provided by the PRA, future plant applications must also address
the technically relevant portions of 10 CFR 50.34(f) and the applicable
reviews discussed in SECY-90-016 (January 12, 1990), "Evolutionary Light Water
Reactor (LWR) Certification Issues and their Relationship to Current
Regulatory Requirements," and SECY-93-087 (April 4, 1997), "Policy, Technical,
and Licensing Issues Pertaining to Evolutionary and Advanced Light Water
Reactor (ALWR) Designs."  The Commission has successfully applied these
requirements as a significant part of the design certification reviews for the
ABWR and the System 80+ designs.  The AP-600 design is undergoing a similar
review process at this time.  The Commission expects that if licensees
reference a certified design, they will maintain the design features that were
included to prevent and mitigate severe accident risk.

Much of the current Commission activity in the area of severe accidents is
being coordinated under the PRA Implementation Plan, for which the latest
quarterly report is SECY-97-076 dated April 3, 1997.  A major ongoing program
included in the PRA Implementation Plan is that of the development of
risk-informed, performance-based regulations for operating plants (Direction
Setting Issue DSI-12).  General guidance for risk-informed activities has been
developed and issued for public comment.  These general documents are the
draft Regulatory Guide DG-1061,"An Approach for Using Probabilistic Risk
Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current
Licensing Basis"; its companion Standard Review Plan, "Use of Probabilistic
Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General
Guidance, Draft SRP Chapter 19, Revision L"; and draft NUREG-1602, "The Use of
PRA in Risk-Informed Applications."  Also, a series of draft
application-specific regulatory guides and standard review plans addressing the topics of
inservice testing, plant technical specifications and graded quality assurance

have been developed and issued for public comment. Similar documents for inservice inspection are currently being prepared for comment. When approved, these documents will provide a framework for future consideration of risk-informed regulatory activities. In addition to the development of these risk-informed guidance documents, licensee pilot plant applications demonstrating the use of this new approach to risk-informed inservice testing, graded quality assurance and technical specifications are currently under review by the staff.

The discussion above provides a general perspective on how the Commission's consideration of risk from severe accidents has evolved over time. Attachment 1 provides additional information on Commission policy statements, regulatory decisions, and other actions that involved consideration of severe accident risk. It is emphasized that the discussion in this paper and the items in attachment 1 are not intended to be all inclusive, but rather to provide a summary of the evolution of the Commission's consideration of severe accident risk leading to current practice.

CONCLUSION:

The Commission has historically considered severe accident risk in making regulatory decisions. The degree to which severe accident considerations have affected the Commission's regulatory activities has increased regularly and substantially over time both in scope and in level of sophistication as improved information about severe accident risk has been developed. This includes information on the frequency of severe accidents as well as their consequences. As more information has become available, additional insights have enhanced the ability of the Commission to make risk-informed decisions. The Commission's safety goal policy and regulatory analysis guidelines have played a strong role in developing requirements to address severe accident risk.

For future regulatory decisions and actions, the staff recommends that

the
Commission continue its current practice of considering severe accident risk
as appropriate in assessing safety issues and the need for regulatory action
either on a generic or a plant-specific basis. More specifically, the continuing application of the Backfit Rule (Part 50.109), the guidance provided in the Severe Accident Policy Statement, and the ongoing risk-informed regulatory development effort under DSI-12 will provide the staff
with guidance for addressing severe accident issues in the future. The staff
also believes that continued consideration of the Commission safety goals along with cost/benefit consideration is appropriate in assessing the need for
future regulatory actions. (Note that in response to an SRM dated July 2,
1997, the staff is preparing an evaluation and recommendation regarding the
merits of elevating the subsidiary core damage frequency (CDF) goal of 1x10-4
per reactor year to the status of a fundamental safety goal.) Based on continuing research at the NRC and in other countries, the knowledge base on
the probability and consequences of severe accidents will continue to increase. This will lead to improved understanding of severe accident phenomenology that will improve the quality of future regulatory decision making. In a separate Commission paper, the staff has provided a recommendation to the Commission regarding generic rulemaking on severe accidents for future light water reactors (SECY-97-148).

COORDINATION: OGC has reviewed this paper and has no legal objection.

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for Operations

Attachment: Commission Consideration of Severe Accident Risk

ATTACHMENT

COMMISSION CONSIDERATION OF SEVERE ACCIDENT RISK

Updated Source Term. In 1962 the U.S. Atomic Energy Commission published TID-14844, "Calculation of Distance Factors for Power and Test Reactors" which
specified a release of fission products from the core to the reactor containment in the event of a postulated accident involving a "substantial
meltdown of the core." This "source term," the basis for the NRC's Regulatory
Guides 1.3 and 1.4, has been used to determine compliance with the NRC's reactor site criteria, 10 CFR Part 100, and to evaluate other important plant
performance requirements. During the past 30 years substantial

additional information on fission product releases has been developed based on significant severe accident research.

As a result of this research, a revised accident source term has been developed for regulatory applications for future LWRs (NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995). Insights from severe accident research on fission product release and transport were used in developing the revised source term. The revised source term is expressed in terms of times and rates of appearance of radioactive fission products into the containment, the types and quantities of the species released, and other important attributes such as the chemical forms of iodine, given a severe core-melt accident. This mechanistic approach provides, for regulatory purposes, a more realistic estimate of the amount of fission products present in the containment from a postulated severe accident than was included in TID-14844. This source term can have implications on issues such as Part 100 reactor siting criteria, equipment qualification, control room habitability, and assessments of severe accident risks in plant environmental impact statements.

NEPA and Classification of Postulated Accidents The National Environmental Policy Act (NEPA) was passed by Congress in December 1969. The initial response to this law by the Atomic Energy Commission (AEC) was criticized by environmentalists as being too narrow, and the Calvert Cliff's Decision in July 1971 by the U.S. Court of Appeals for the District of Columbia required the AEC to broaden its scope in evaluating potential environmental impacts of nuclear power plants. In this connection, in December 1971, the AEC proposed that a rulemaking be held to amend Appendix D of 10 CFR Part 50 to include an "Interim Statement of General Policy and Procedure: Implementation of the National Environmental Policy Act of 1969." The proposed Annex to Appendix D specified certain standardized assumptions to be made in evaluating risks due to postulated accidents in environmental reports submitted by applicants for construction permits or operating licenses for nuclear power reactors. Since it was not practical to consider all possible accidents, a scheme was devised for classifying accidents into a spectrum of potential accidents ranging in severity from trivial (Class 1) to very serious accidents (Class 9) involving core meltdown. Classes 1-8 were to be addressed by applicants in their environmental reports for review by the AEC. Class 9 accidents were not required to be analyzed due to their perceived low likelihood. The proposed Annex was formally withdrawn and the rulemaking suspended in June 1980. Among the reasons given for the withdrawal were that the Annex did not properly prescribe attention to the kinds of accidents (Class 9) that dominated accident risk, the definition of Class 9 accidents was imprecise, the prescriptions of assumptions to be used in environmental analysis did not contribute to objective consideration, and inadequate consideration was given to the prevention and mitigation of accidents.

Severe Accident Management Design Alternatives (SAMDAs). The U.S. Court of Appeals, in Limerick Ecology Action v. NRC, 869 F.2d 719 (3d Cir. 1989), effectively required the NRC to include consideration of certain SAMDAs in the environmental impact review performed as part of the operating license application for the Limerick Generation Station. The review of SAMDAs for Limerick was published as a Supplement to NUREG-0794, "Final Environmental Statement Related to the Operation of Limerick Generation Station, Units 1 and 2," dated August 1989. Subsequent to the Limerick review, SAMDAs have also been considered and documented in a Supplement to NUREG-0775, "Final Environmental Statement Related to the Operation of Comanche Peak Steam Electric Station, Units 1 and 2," dated October 1989.

The purpose of the requirement to consider SAMDAs in the environmental impact reviews was to ensure that plant design changes with the potential for improving severe accident safety performance were recognized and evaluated. For example, the staff assessed TVA's SAMDA evaluation for Watts Bar, Unit 1. TVA had identified a set of potential SAMDAs for Watts Bar through a systematic assessment of the key contributors to risk at the plant. Quantitative estimates of risk reduction associated with potential design improvements were developed based on the PRA. The risk reduction potential for each enhancement was based on calculating the change in the core damage frequency (CDF) and total risk. This calculation, along with the cost impact of candidate design improvements, was used to determine a cost/benefit ratio for each enhancement. A systematic screening criterion considering the estimated cost per person-rem averted for the various SAMDAs (including the impact of uncertainties in the averted offsite risk estimates) was used to evaluate which design improvements warranted implementation at Watts Bar.

Subpart B of Part 52 of Title 10 of the Code of Federal Regulations (10 CFR Part 52) does not specifically require an environmental impact statement (EIS) for a standard plant design certification. However, a NEPA evaluation in the form of an EIS that considers severe accident mitigation design alternatives is an essential element of an application for a combined license under Subpart C of 10 CFR Part 52, for those applications that reference a design certified under Subpart B. Therefore, a cost-benefit analysis of various design alternatives related to the prevention and mitigation of severe accidents, similar to that described above, is included as part of the design certification rulemaking for standard designs.

Anticipated Transients Without Scram (ATWS). The Commission issued requirements to reduce the risk from ATWS events for PWRs and BWRs (10 CFR 50.62). ATWS accidents had been a concern because under certain postulated conditions they could lead to severe core damage and release of radioactivity to the environment. In promulgating the ATWS rule, the Commission stated that this new regulation would "significantly reduce the risk of nuclear power plant operation." The staff prepared a regulatory analysis for the ATWS rule in which they used PRA information to evaluate the costs and values (to estimate the value/impact ratio) of various alternatives for implementing the new ATWS requirements.

The estimated benefit from implementing the rule was a reduction in the frequency of core damage per reactor-year due to ATWS and the associated reduction in risk to the public from accidental release of radioactive material. Thus, severe accidents were considered in promulgating the ATWS rule in order to reduce the risk from a postulated accident beyond the design basis. These remarks apply equally to the station blackout rule in the following discussion.

Station Blackout Rule. Station blackout (SBO) involves the concurrent

failure
of both offsite and onsite emergency AC power supplies. This condition represents an accident beyond the normal design basis. In 1975, the results
of the Reactor Safety Study (WASH-1400) showed that station blackout could be
an important contributor to the total risk from nuclear power plant accidents.
Subsequent technical evaluations and risk studies showed that no undue risk
existed with or without promulgation of the station blackout rule. However,
station blackout could still be an important contributor to residual risk.
Therefore, the Commission issued the Station Blackout Rule (10 CFR 50.63) to
enhance safety by accident prevention and thereby reduce the likelihood of a
core damage accident caused by a station blackout event. IPE results from
draft NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," dated November 1996, indicate that significant reduction in CDF is achievable through the implementation of the
SBO rule. For 15 plants, including both PWRs and BWRs, for which risk reduction values were provided, the average value of CDF reduction was reported to be 2x10-5 per reactor year.

Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants. This policy statement (April 1983) describes how the Commission intended to resolve safety issues related to reactor accidents more
severe than design basis accidents. The focus is on guidance for regulatory
decision making on how severe accident issues should be treated for existing
and future nuclear reactors, with special focus on certification of new standard plant designs. Although the Commission concluded that existing plants do not pose an undue level of risk to the public, the Commission expects new standard plants will achieve a higher standard of severe accident
safety performance than prior designs.

The expectation that new designs can achieve a higher standard of severe accident safety performance is based on the growing information that has come
from research and operating reactor experience that has improved our knowledge
of specific severe accident vulnerabilities and cost-effective methods for
their mitigation. Realistic evaluations of core-melt accidents and potential
containment failure are expected to be performed for these designs taking into

account severe accident phenomena such as dynamic and static loading from combustion of hydrogen and other combustibles, static pressure and temperature loadings from steam and non-condensibles, basemat penetration by core-melt materials, and effects on aerosols on engineered safety features.

For existing plants, the Commission policy stated that no further regulatory actions to deal with severe accident issues are required unless significant new safety information arises to question the conclusion that existing plants pose no undue risk to public health and safety.  To verify this conclusion, licensees of each operating reactor were expected to perform an accident safety analysis designed to discover instances of particular vulnerability to core melt or to unusually poor containment performance given a core melt (See IPE discussion below).

With regard to future reactors, the Commission determined that, for new designs to demonstrate acceptability regarding severe accident concerns, they must undergo a Probabilistic Risk Assessment (PRA) to evaluate potential severe accident vulnerabilities and to develop insights into design-specific plant behavior under severe accident conditions.

Integration Plan for Closure of Severe Accident Issues.  In 1988 the staff sent a plan to the Commission for closure and integration of severe accident issues (SECY-88-147, May 25, 1988).  This plan provided a coordinated effort to ensure fulfillment of the Commission's Severe Accident Policy Statement.
The six main elements of the plan are: (1) the individual plant examination (IPE) program, (2) a containment performance improvements (CPI) program for each of the six containment types, (3) a program to improve plant operations, (4) a severe accident research program (SARP), (5) an external events program, and (6) an accident management program.  Completion of the elements of this plan would constitute a basis to ensure that the residual risks to the public from severe accidents at nuclear power plants are minimized in an effective manner.  Each year the staff informs the Commission on the status and

progress in implementing the elements of the integration plan. The latest update was provided to the Commission (SECY-97-132, June 23, 1997). Therefore, that information will not be repeated in this paper.

There were regulatory decisions that have been made in which severe accident risk was an important factor in the decision-making process. The CPI program assessed generic severe accident challenges to each LWR containment type to determine whether additional regulatory guidance or requirements concerning needed containment features are warranted. Such assessments were deemed necessary at the time because of the relatively large uncertainty in the ability of LWR containments to successfully survive some severe accident challenges. For each containment type, a number of generic potential containment and plant improvements were evaluated to determine the potential benefits in terms of reducing the core melt frequency, containment failure probability, and offsite consequences. A cost-benefit analysis was done to determine the priorities and recommendations for the various alternatives. Based on the results of this analysis, the staff issued a generic letter (Generic Letter 89-16) to licensees with Mark I containments requesting that hardened vents for containment pressure relief capability be installed. Subsequently, all operating MARK I plants installed hardened vents. Although no generic improvements were identified for the other containment types, a number of insights were identified that were provided to licensees (Generic Letter 88-20, Supplement 3) for use in their Individual Plant Examinations.

Following SECY-88-147, the staff issued a revised SARP Plan (NUREG-1365, August 1989). A significant portion of the revised SARP was directed toward issues that related to major areas in the Integration Plan. In particular, issues and accident sequences that lead to potential early containment failure (e.g., direct containment heating and BWR Mark I containment shell meltthrough) were the focus of near-term research because these issues were considered to be of high risk significance. In 1992 the staff issued an update to the SARP Plan (NUREG-1365, Rev. 1, December 1992). Among other things, this update identified the near-term severe accident issues that were closed or were near completion and described the progress in understanding other important severe accident phenomena.

Significant efforts have been applied to assess the risk from, and the likelihood of, potential early containment failure in the event of a severe accident. Two of these issues have been resolved: early failure of the Mark I containment due to direct contact between core debris and the containment, and the alpha-mode (steam explosion) containment failure. For the issue of containment attack from core debris, it was concluded that if water is assumed to overlie the molten core material as it spreads on the drywell floor toward the containment liner, containment failure would be physically unreasonable. In the absence of water, however, it was concluded that the containment barrier would be failed. This information was provided to licensees for their consideration in developing accident management procedures. As a result, the BWR owners' group prepared a document entitled "Emergency Procedures and Severe Accident Guidelines" that increases the priority for using drywell sprays to provide water to the drywell to prevent liner meltthrough in the event of a severe accident. The staff will continue to work with the Owners' Group to ensure that the final guidelines are consistent with the technical conclusions of the liner meltthrough issue.

The alpha-mode failure of the containment (steam explosion) was identified in WASH-1400 as a potentially important contributor to early containment failure. Alpha-mode failure was postulated to occur as a result of an in-vessel steam explosion that produces a missile that could subsequently result in containment failure. This mode of containment failure was also evaluated during the IPE reviews for certain plants. However, it has been concluded that the overall likelihood of early failure from this challenge is low. Alpha-mode failure was also evaluated in NUREG-1150 and was determined to have a likelihood too low to be an important severe accident issue. In June 1995, the Second Steam Explosion Review Group Workshop (SERG-2) was held to review the status of fuel-coolant interaction research. The results of this review meeting were published in NUREG-1524, "A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of the Current Understanding of Broader Fuel-Coolant Interaction Issues," in August 1996. The overall conclusion of the majority of the international experts participating in SERG-2, was that alpha-mode failure was a very low probability event and therefore resolved from a risk perspective.

Direct containment heating (DCH) was identified as one of the important contributors to early containment failure for PWRs in NUREG-1150 and in the IPEs. DCH refers to the process whereby under certain accident scenarios, molten core debris is ejected under high pressure from the reactor vessel into the containment atmosphere. The subsequent rapid heating of the containment atmosphere, in conjunction with possible hydrogen combustion, can lead to early containment failure. The staff has completed a significant portion of its evaluation of DCH which involved a substantial amount of testing and analysis. The results indicate that for 41 Westinghouse large, dry and subatmospheric containment reactors, DCH poses no tangible threat to containment integrity. The resolution of this issue for a substantial number of plants eliminates this matter from further analysis. (It should be noted that several concerns have been raised by an individual regarding the resolution of DCH. These concerns are being reviewed by the staff.) Additional work is under way to resolve this issue for the remaining PWR plants.

Policy Statement on Safety Goals for the Operations of Nuclear Power Plants.

The Commission's policy statement on safety goals (August 1986) focuses on risks to the public from the release of radioactive materials to the environment for normal operations as well as from accidents. The Commission established two qualitative safety goals which are supported by two quantitative objectives. Based on the quantitative objectives, the staff is using a subsidiary safety goal objective for core damage frequency (CDF) of 1x10-4 per reactor year for screening purposes in prioritizing regulatory activities and for making comparisons between predicted plant performance under severe accident conditions and the Commission's safety goals. The staff is also proposing the use of a CDF guideline of 1x10-4 per reactor year for use in risk-informed decision making along with a large early release frequency (LERF) of 1x10-5 per reactor year.

In developing this policy statement, the Commission considered that severe core damage accidents can lead to more serious accidents with the potential for life-threatening offsite release of radiation, for evacuation of members of the public, and for contamination of public property. In order to avoid these adverse consequences, the Commission also stated that it will continue to pursue a regulatory program that has an objective of providing reasonable assurance (while giving appropriate consideration to the uncertainties involved) that a severe core damage accident will not occur at a U.S. nuclear power plant. A number of uncertainties (e.g., thermal-hydraulic assumptions and the phenomenology of core melt progression, fission product release and transport, and containment loads and performance) arise because of a lack of severe accident experience or detailed knowledge of accident phenomenology along with data related to probability distributions. However, sensitivity studies can be performed to determine which of these parameters are most important to probabilistic estimates. Sensitivity studies such as these have been used during the design certification reviews of the ABWR, System 80+ and AP600 advanced ALWR designs.

Prioritization of Generic Safety Issues. After TMI, many new generic safety issues were raised. The TMI-2 action plan called for early resolution of generic safety issues. Prior to TMI, a methodology was developed to prioritize issues based on a quantitative estimate of the risk reduction associated with the potential change in requirements that could result from resolving an issue and the estimated costs to implement such a change. After TMI, risk-informed prioritization was further developed and utilized as an input to generic regulatory decisions involving the use of resources by licensees. The primary purpose of prioritization is to assist in the timely and efficient allocation of resources to those generic safety issues that have a high risk implication. The methodology and criteria for assigning priorities are documented in NUREG-0933, "A Prioritization of Generic Safety Issues." High, medium, low, or drop priority rankings for each issue are determined based on the estimated impact/value ratio (dollars per person-Rem) and change in core damage frequency per reactor-year associated with the issue. The value, or risk reduction estimates, are based on the expected reduction in radiological consequences that the resolution could effect.

Regulatory Analysis. The NRC performs regulatory analyses for all proposed new requirements. This analysis includes an assessment of the value and impacts of the proposed actions (e.g., rules, bulletins regulatory guides) by demonstrating that a substantial increase in the overall protection of the public health and safety is justified in light of the costs for implementing the new requirement. NUREG/BR-0058, Rev. 2, "Regulatory Analysis Guidelines of the U.S. NRC," provides guidance on performing regulatory analyses. Severe accident risk is considered in these regulatory analyses. Part of the regulatory analysis includes a safety goal evaluation in which changes in the estimated core damage frequency per reactor-year are considered in addition to the estimated conditional probability of early containment failure or containment bypass. NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," provides more details on the preparation of regulatory analyses to aid NRC in deciding whether or not a proposed new regulatory requirement should be imposed. This report includes a discussion on the safety goal evaluation as well as detailed guidance on the performance of the value-impact analysis portion of the regulatory analysis. An important part of the value part of the equation is an estimate of the expected change in radiation exposure to the public due to changes in accident frequencies or consequences associated with the proposed action. Improvement in the state of knowledge for factors such as accident frequencies or consequences can ultimately lead to a reduction in uncertainty.

Individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities. In the Commission policy statement on severe accidents in nuclear power plants, the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on generic rulemaking or other regulatory requirements for these plants. However, the Commission recognized, based on NRC and industry experience with plant-specific PRAs, that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements. Therefore, in November 1988, the staff issued Generic Letter 88-20 that requested each existing plant to perform a systematic examination (i.e., IPE) to identify any plant-specific vulnerabilities to severe accidents.

The general purpose of this examination was for each utility (1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant, (3) to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (4) if necessary, to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

In addition to internal events, risk assessments had also indicated that the risk from external events could be a significant contributor to core damage in some instances. Therefore, in 1991, the staff issued Supplement 4 to Generic Letter 88-20 that requested each utility to perform a systematic individual plant examination for severe accidents initiated by external events (IPEEE). The general purpose of the IPEEE was similar to that of the internal event IPE that was requested in Generic Letter 88-20. Utilities were requested to submit the results of their IPE and IPEEE for each plant to the NRC.

Policy Statement on the Regulation of Advanced Nuclear Power Plants. This policy statement (July 8, 1986) states that advanced reactors are expected to provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions. Features should be considered in advanced designs that minimize the potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity, and independence in safety systems.

Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Activities (August 1995). Using information from PRAs enhances the traditional deterministic approach to regulation by providing a logical means to prioritize potential challenges to safety based on their risk significance. Significant improvements in PRA techniques (e.g., NUREG-1150), as well as the results of a substantial research program on severe accident phenomenology since TMI, enabled the Commission to greatly improve its methods for assessing containment performance after a core-damage accident. Therefore, the Commission issued a policy statement to expand the use of PRA in all regulatory matters as a complement to the NRC's deterministic approach and defense-in-depth philosophy. In the policy statement it is stated that the objectives of the use of a probabilistic approach in regulation are: (1) to allow consideration of a broader set of potential challenges to safety, (2) to provide a logical means for prioritizing these challenges based on risk significance, and (3) to allow consideration of a broader set of resources to defend against these challenges.

PRA Implementation Plan. In parallel with the publication of the final policy statement, the staff developed an implementation plan to define and organize the PRA-related activities being undertaken. Each quarter the staff provides an update to the Commission on the progress of activities in the PRA Implementation Plan. (The most recent quarterly update was SECY-97-076, dated April 3, 1997.) These activities cover a wide range of PRA applications and involve the use of a variety of PRA methods. For example, applications involve the use of PRA in the assessment of operational events in reactors; developing guidance for NRC inspectors on focusing inspection resources on risk-important equipment; and regulatory guides for inservice testing, graded quality assurance, and changes to plant technical specifications. Key issues that are being addressed in developing this guidance relate to risk-informed decision making, in particular, criteria to allow changes to overall plant risk. SECY-97-077, dated April 8, 1997, provided the Commission with four draft Regulatory Guides, three draft Standard Review Plan sections, and draft NUREG-1602, "The Use of PRA in Risk-Informed Applications," that support implementation of risk-informed regulation for power reactors.

One of the four draft regulatory guides that has been released for public comment describes the general approach for using PRA in making risk-informed decisions on plant-specific changes to an operating plant's current licensing basis (CLB). This general approach is documented in Draft Regulatory Guide DG-1061 (Attachment 2 to SECY-97-077). In this draft document, guidance is provided on using risk information in support of licensee-initiated CLB changes that require review and approval by the NRC. This effort is a part of the activities associated with Direction Setting Issue 12, Operating Reactor Program Oversight. Consideration is explicitly given on assessing the impact that proposed changes to the plant's CLB have on the risk associated with the plant's design and operation. One of the principles used in implementing this risk-informed decision making is that proposed increases in risk should be small and should not cause the NRC safety goals to be exceeded. Core damage frequency (CDF) and large early release frequency (LERF) are proposed as measures for making risk-informed regulatory decisions. Therefore, increases in CDF and LERF resulting from proposed CLB changes are limited to small increments. Regulatory Guide DG-1061 includes acceptance guidelines for various combinations of initial (baseline) plant CDF and LERF and calculated changes in these values expected to result from the implementation of a proposed risk-informed, performance-based change in plant operation. Results of plant-specific PRAs are compared with the acceptance guidelines taking into account significant model uncertainties in PRAs including the phenomenology of accident progression and mechanisms for the release of fission products.