UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 7 1983

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Ahearne
Commissioner Roberts
Commissioner Asselstine

FROM: William J. Dircks, Executive Director for Operations

SUBJECT: MODIFICATIONS TO SEVERE ACCIDENT POLICY STATEMENT (SECY-82-1B)

The staff has updated the "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation" as a result of review comments from the following sources:

(1) The changes asked for in the notation votes received from Chairman Palladino and Commissioner Ahearne;

(2) The changes proposed by the staff and presented at the Commission meeting on SECY-82-1B on January 6, 1983 (see pp. 3 and 12);

(3) Changes in Part IX resulting from items 4a, b and c of the ACRS letter to Chairman Palladino dated January 10, 1983; and,

(4) The changes proposed by Dade Moeller of the ACRS as presented to the staff at the ACRS meeting on December 21, 1982.

Enclosure 1 is a comparative copy showing deletions from and additions to the proposed policy statement in SECY-82-1B. The additions are underlined and the deletions are dashed through. There are minor editorial changes that have been made in the proposed policy statement to improve clarity. These changes are not marked in the enclosure.

Our purpose in transmitting this update of the proposed statement is to aid the discussion on February 10 when the Commission, ACRS and staff meet to reconcile the remaining differences of views on this policy matter.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

Enclosure:
Modified SECY-82-1B

cc: ACRS (20)
NRR
RES
SECY
OPE
OGC

CONTACT:
Roger J. Mattson
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8302170132 XA

A Copy Has Been Sent to PDR

ENCLOSURE 1 (Comparative copy with deletions from and additions to SECY-82-1B as of 2-2-83)

[7590-01]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of proposed Commission Policy Statement.

SUMMARY: This proposed Policy Statement summarizes the changes in rules, policies, and regulatory practices that constitute the NRC approach for severe accident rulemaking. The new approach as presented in the Policy Statement would, for all classes of existing or proposed nuclear power plants, replace unfocused, long-term generic rulemaking with (1) severe accident rulemakings designed to certify future standard plant designs and (2) regulatory decisions based on generic evaluations and decisions regarding all existing or proposed plants for which the standard plant rulemaking would not apply. The Policy Statement is presented in proposed form to provide all affected nuclear power plant licensees and applicants and other interested persons an opportunity to comment.

DATES: Submit comments by [insert date - 90 days from publication]. Comments received after that date will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before this date.

ADDRESSES: Submit comments, suggestions, or recommendations to the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Branch. Copies of comments received may be examined in the NRC Public Document Room, 1717 H Street, NW., Washington, D. C.

FOR FURTHER INFORMATION CONTACT: Roger J. Mattson, Director, Division of Systems Integration, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, telephone (301) 492-7373.

SUPPLEMENTARY INFORMATION: The U. S. Nuclear Regulatory Commission published "The Advanced Notice of Rulemaking" in the Federal Register on October 2, 1980 [45 FR 65474]. In that notice, the Commission indicated that a long-term rulemaking effort was being initiated that would establish policy, goals, and requirements relating to core-melt accidents greater than the present design basis accident and invited public comment on proposals for treating severe accident issues. This Policy Statement summarizes the changes in rules, policies, and regulatory practices that constitute the NRC approach for severe accident rulemaking. The new approach would, for all classes of reactors, replace an unfocused, long-term generic rulemaking effort with severe accident rulemakings designed to certify specific standard plant design applications and with regulatory decisions based on generic evaluations and decisions regarding other classes of operating plants, ~~or~~ plants under construction or proposed plants. It is expected that this approach would fully resolve the severe accident safety issues in the course of these rulemakings on specific standard plant designs and regulatory decisions on other classes of existing or future plants which may, or may not, include rulemaking. The Policy Statement proposes that final decisions on severe accident considerations for operating plants and plants under construction be accomplished in parallel with the standard plant reviews.

PROPOSED COMMISSION POLICY STATEMENT

ON

SEVERE ACCIDENTS AND RELATED VIEWS ON NUCLEAR REACTOR REGULATION

I. INTRODUCTION: HISTORY AND PURPOSE OF THE POLICY STATEMENT

The Nuclear Regulatory Commission mandated a series of changes in design and operation of nuclear power plants as a response to deficiencies revealed by the accident at Three Mile Island (TMI). The changes began with the operating Babcock and Wilcox plants and then the other operating plants. Later, the Commission set requirements for plants whose operating license (OL) review had been interrupted by the attention paid to operating plants. Still later, a separate set of requirements was developed for plants whose construction permit (CP) review had been interrupted. This last set of requirements, embodied in the Construction Permit/Manufacturing License Rule (hereinafter, the CP Rule) was published in effective form on January 15, 1982 (47 FR 2286).

In this Policy Statement, the Commission describes its policy and requirements for new CP applications and reactivated CP applications, and the Commission reiterates and discusses its present requirements with respect to accidents more severe than design basis accidents. We connect all of these requirements to our "Proposed Policy Statement on Safety Goals for Nuclear Power Plants" (47 FR 7023, February 17, 1982); to our standardization rules; and to other policy guidance under development such as siting policy. Although gas-cooled or other types of advanced reactors may be proposed in the future, they have not been considered in the development of this policy statement. Some of the policy points would apply to such plants; others would not.

As part of the Commission's response to TMI, an Action Plan (NUREG-0660, May 1960) was issued. Section II.B of that plan deals with the siting of plants and the requirements for coping with severe accidents. Consistent with that plan, the Commission has already issued one final and one proposed interim rule concerning hydrogen control issues in degraded core cooling (46 FR 58484, December 2, 1981, and 46 FR 62281, December 23, 1981). The concept of a generic rulemaking to reach final decisions on severe accidents also took form in the TMI Action Plan, Task II.B.8, "Rulemaking Proceeding on Degraded Core Accidents." This plan envisioned a long-term rulemaking extending beyond 1982 to establish policy, goals, and requirements related to accidents involving core damage greater than the present design basis for all classes of reactors: those operating, under construction, proposed for construction, or proposed as new standard plant designs. The task also included the interim step of an Advanced Notice of Rulemaking, issued on October 2, 1980 (45 FR 65474).

The presently proposed Policy Statement replaces this advanced notice of rulemaking. It represents a change from the envisioned plan for long-term rulemaking covering all classes of nuclear power plants in that the focus of rulemaking would, if adopted, be reduced to only one class of plants, namely, those proposed as new standard plant designs. However, the proposed Policy Statement provides the current views of the Commission on the process for arriving at severe accident decisions for operating plants, those under construction, or proposed for construction for which standard plant rulemaking would not apply.

For the reasons discussed below, the Commission believes that ~~existing~~ nuclear power plants of modern designs (such as those proposed in CP applications docketed after the promulgation of the Standard Review Plan or now under consideration by U.S. vendors for future sales) can be shown to be acceptable for severe accident concerns if they meet the requirements of the CP rule; if they achieve a technical resolution of Unresolved Safety Issues; and if they are adequately responsive to insights afforded by probabilistic risk assessments. This conclusion embodies due consideration of the Commission's proposed policy statement on safety goals. It permits plants of modern design to be sited at locations with demographic and other safety-related characteristics that conform to our siting regulations and guidance. Further discussion of siting policy revision is found in Section IX of this Policy Statement.

As discussed below, our policy for the consideration of severe accidents contains nine interrelated components: (1) proposed Policy Statement on safety goals; (2) use of probabilistic risk analysis in severe accident decisionmaking; (3) lessons learned from TMI; (4) the Standard Review Plan; (5) standardization policy; (6) further research on severe accidents; (7) treatment of severe accidents in ongoing licensing proceedings; (8) present views on other safety issues and efforts in progress; and (9) implementation of severe accident policy.

In accordance with the activities, views, and policy developments discussed in this Policy Statement, the Commission believes that it is possible to begin reviews of specific standard plant design applications with an expectation of fully resolving the severe accident questions in the course of the review. This belief is predicated on the availability of results from ongoing NRC, Industry Degraded Core Rulemaking (IDCOR), and vendor research; and insights from the Zion, Indian Point, and other risk assessments. The review of standard designs for future CPs provides incentive to industry to address severe accident phenomena. These reviews and ongoing research will also provide information needed for final decisions on severe accident considerations for operating plants and plants under construction. We expect to reach those final decision within the next several years.

A three-step process will be used for severe accident decisions for plants in operation, under construction, or other classes of plants proposed for construction for which standard plant rulemaking would not apply. First, we will broadly assess the safety of these plants using probabilistic risk assessment. We do not plan for additional PRAs to be generated for this purpose. The existing ensemble of available probabilistic risk assessments (presently about 13 in number), will be normalized by updating of accident likelihood predictions and by recalculations of accident consequences using revised source terms currently being evaluated by the NRC staff. The ensemble of PRAs will provide benchmarks for estimating the risk of other existing LWR plants. This approach will provide better understanding of the design features and site characteristics that are more favorable or less favorable to various risk contributions relative to the plants in the ensemble. This, in turn, will provide an envelope of risk for the various accident scenarios that are the dominant contributors to severe accident risk and it can also provide benchmarks for judgmental comparisons of design changes. Second, a range of possible design and operational changes to improve accident prevention and consequence mitigation capabilities will be studied to determine the costs and safety benefits of backfitting them to plants in operation or under construction. Finally, using a Commission safety goal (if approved in the future for this application) in combination with engineering and policy judgment, decisions will be made on whether ~~improvements~~ reductions in severe accident risk are necessary ~~or possible~~. If ~~improvements~~ reductions are necessary, our research should tell us how best to achieve them. We will also be able to decide whether the costs for various safety improvements are justified.

Our current general licensing policy and outline of future activities and schedules for severe accidents are treated in more detail below. Especially important for decisions on operating plants and plants under construction is their connection with severe accident research as described below.

II. POLICY ON SAFETY GOALS

The Commission published their policy statement on safety goals for the operation of nuclear power plants in February, 1983. The policy statement contains qualitative safety goals and quantitative design objectives which are intended to be consistent with the qualitative goals. The Commission also announced the start of a two-year period of evaluation for the policy statement in February, 1983. During the evaluation period, the qualitative safety goals and quantitative design objectives will not be used in the licensing process or be interpreted as requiring the performance of probabilistic risk assessments by applicants or licensees (see Part III). Rather, the NRC will continue to use conformance to regulatory requirements as the exclusive licensing basis for plants. Use of the policy statement during the evaluation period will be limited to uses such as examining proposed and existing regulatory requirements, establishing research priorities (see Part VII), resolving generic issues (see Parts IX, X), and defining the relative importance of issues as they arise. At the conclusion of the evaluation period, the Commission will consider if any revisions are necessary before the issuance of a final policy statement and a plan for its implementation.

(See page 6a for original text of Part II)

III. USE OF PROBABILISTIC RISK ASSESSMENT IN SEVERE ACCIDENT DECISIONMAKING

Probabilistic risk assessment is a process that can be used for reviewing design and operation of a nuclear power plant. It provides an integrated assessment of the relative importance of potential accident sequences and ~~identifies~~ helps identify the weaknesses in plant design and operation that contribute to the most important accident sequences. Many PRAs of U.S. nuclear power plants have been made since two plants were analyzed and reported in the Reactor Safety Study (WASH-1400). These PRAs include risk

II. ~~PROPOSED~~ POLICY ON SAFETY GOALS

On February 17, 1982, the Commission published for comment a proposed policy statement on safety goals for nuclear power plants, which presents several qualitative safety goals and quantitative probabilistic guidelines for severe nuclear accidents (47 FR 7023). It also includes numerical guidance to ensure that the risks of nuclear power plants are as low as reasonably achievable. The Commission is presently revising the safety goals policy statement, taking into consideration public comment. It is anticipated that a final statement on safety goals will be issued for trial use. During the trial use period, conformance with the safety goals and associated numerical guidance will not be required except as directed by the Commission. It is anticipated that during the trial use period, the Commission will require comparisons of proposed new generic requirements with the safety goal prior to adopting such new requirements, whether for severe accident considerations or otherwise.

assessments done under the NRC's Reactor Safety Study Methodology Application Program (RSSMAP) and the Interim Reliability Evaluation Program (IREP), as well as a number of industry studies (Big Rock Point, Limerick, Zion, and Indian Point).

A continuation of IREP has been designated the National Reliability Evaluation Program (NREP). In the future, NREP may be implemented on other operating plants within the United States, individually or in groups, using a standard methodology emanating from IREP or the NRC and industry forum on PRA procedures sponsored by the Institute of Electrical and Electronics Engineers (IEEE) and the American Nuclear Society (ANS).

Thus far, the PRAs of nuclear power plants have varied in scope, depth, and quality; but, taken as a whole, they indicate measurable growth in the constructive use of the techniques of PRA to develop supplements to current regulatory practice. They lead us to conclude that PRA improves our understanding of the severe accident sequences to which plants are most vulnerable and provides tentative measures of the overall risk posed by specific plants (as well as the dominant constituents of that risk). In sum, considering the experience with risk assessments thus far made, we conclude that the cost-effectiveness of risk reduction measures can be studied through PRA. Although there are limitations due to the many uncertainties associated with the use of PRA, the Commission considers it to be a valuable adjunct to the established regulatory process and NRC's reactor safety regulations in 10 CFR, Chapter I.

Some of the previous risk assessments have identified new equipment that, if added, and specific plant features that, if modified, have a high potential for risk reduction. Such features typically involve details of system design and operation and not the more fundamental and costly aspects of design. Some examples of details of system design and operation are discussed in Sections VIII and IX of this Policy Statement.

It is our judgment that the utility of PRA can be improved if it is integrated with the design process. To take advantage of this improved use of PRA, the Commission will require the performance of a PRA that is as complete as practical for any standardized design to be referenced in future CP applications. The purpose of these PRAs is twofold: to encourage the development of an effective reliability and risk management program beginning at the design stage and to determine if there should be additional regulatory requirements imposed because of insight gained from the PRA before issuance of a license referencing that design. We believe that such studies can help to identify design features that would lessen the likelihood of degraded-core-cooling events (accident prevention), arrest the extent of damage by successful interdiction of a degraded-core-cooling event (accident management), or lessen the ensuing consequences of a core meltdown (consequence mitigation). We expect that PRA can help to illuminate those design requirements that are practical and that can make a significant, cost-effective contribution to risk reduction. Our regulations already require for nea[illegible]m CPs that PRA be factored into the design process shortly after CP issuance. Thus, our policy for new CP applications referencing standardized designs is simply to require that the PRA and associated reliability engineering programs be performed earlier in both the design and regulatory processes than is now ~~being performed~~ required (the case). The specifics of our standardization policy are discussed more fully in Section VI of this Policy Statement.

We emphasize that PRA is only one of several tools to be used in making backfit decisions for plants already licensed and in developing safety rulemakings for future standardized designs. We also caution that although we intend to encourage broad uses of the PRA methodology in regulatory decisionmaking -- including severe accident analysis in operating nuclear power plants -- we do not expect to develop widespread requirements for compliance with any numerical safety goal ~~guidance~~ design objectives that might be approved for individual licensing reviews until refinements in PRA methodology make it more appropriate for this purpose. Some discussions of provisional numerical guidelines and PRA methodology will emerge in certain licensing hearings where PRAs have been required (e.g., OL applications for plants in high population density sites and new CP applications).

IV. LESSONS LEARNED FROM THREE MILE ISLAND

The lessons learned from TMI have been applied to operating plants, plants in operating license review, and plants now undergoing construction permit review. The lessons are summarized as licensing requirements for operating plants and plants under construction in "Clarification of TMI Action Plan Requirements" (NUREG-0737, November 1980). The Commission's policy for pending CP applicants is that they comply with the CP Rule (47 FR 2286, January 15, 1982). It is our policy that future CP applications or reactivations of CP applications previously docketed also comply with the CP Rule.

Since effective implementation of the actions summarized in NUREG-0737 and the CP Rule have significantly upgraded nuclear power plant safety, a deliberate approach to decisionmaking on severe accidents is warranted. The Commission's policy on licensing requirements for severe accidents prior to final decisionmaking is described in Section X of this Policy Statement.

V. STANDARD REVIEW PLAN

On March 18, 1982, the Commission announced the issuance of a rule (47 FR 11651) that requires future applicants for operating licenses, construction permits, manufacturing licenses, and preliminary or final design approvals for standard plants to identify and evaluate differences from the acceptance criteria of the applicable revision of the Standard Review Plan (SRP) as part of the technical information to be submitted as part of an application. The SRP was originally issued in 1975 with the most recent revision being issued in September 1981 (NUREG-0800).

The SRP describes acceptance bases and criteria for conclusions which are presented in a staff Safety Evaluation Report (SER). Although conformance with the SRP is not a regulatory requirement, the acceptance criteria of the SRP provide for greater stability in the licensing process and, in a growing number of areas, also provide quantitative guidance for ensuring safe performance of a plant. The lessons learned from TMI have also been incorporated into the SRP. Accordingly, the strengthening of the SRP and procedures for its application reduce the urgency for final decisions on severe accidents for plants under construction. Moreover, SRPs provide a useful fiducial for considering new safety requirements for the next generation of plants. Hence, staff review against the SRP is an important part of the network of assurance needed on the acceptability of new plants.

VI. STANDARDIZATION POLICY

On August 31, 1978, the Commission issued a policy statement, "Statement on Standardization of Nuclear Power Plants," that expanded on the standardization concept for nuclear plants and described specific policy changes being made to improve the usefulness of the Commission's standardization program (43 FR 38954). That policy statement, among other things, defines the effective time periods for design approvals under each of the four standardization concepts: (i) the reference system concept; (ii) the duplicate plant concept; (iii) the manufacturing license concept; and (iv) the replicate plant concept.

The Commission reiterates its support for standardization. To this end, holders of, and applicants for, Preliminary or Final Design Approvals should modify their applications to take into account the guidance of this Policy Statement if the design approvals are to be used in future CP applications. The requirements to be met are enumerated in Section X of this Policy Statement. We expect to complete our reviews within a year or two of such applications. In the interim until a severe accident review is completed and a new design approval is granted, a standard design with an approval granted pursuant to present Commission regulations must be updated in order to be referenced in new or reactivated CP applications by showing that it meets the new CP Rule, and an application must be filed for a severe accident review pursuant to the requirements of Section X, below.

When reviewed by the staff and approved in rulemaking, the Commission expects that the approval of the standardized designs for referencing in future CP applications would be binding on both the staff and applicants for a period of ten years unless significant new <u>safety</u> information becomes available. Design changes considered in response to this new information would be reviewed to ensure that risk reduction is cost-effective before initiating further rulemaking to incorporate the changes. Regulatory or national standards issued subsequent to the approval of a standardized design (e.g., later editions of the ASME Code) may be proposed by applicant and used when agreed to by the staff in lieu of or in addition to those referenced if commercial practice makes it desirable.

Ten-year referenceability of approved designs appears to be a reasonable choice in view of the long lead times experienced in the past five to ten years in effecting significant design improvements; further, it is a time span consistent with practical use of standardized designs.

The Commission intends that approval of standard designs in accordance with Section X, below, be accomplished by rulemaking in accordance with 10 CFR Part 50, Appendix O, Section 7. Applicants seeking this course will be given priority over applicants for new custom plant approvals in the assignment of staff review resources.

The Commission acknowledges the importance of having final design information in the performance of probabilistic risk assessments. This may require essentially an FDA-level of design detail for the nuclear steam supply system (NSSS) and for a substantial portion of the balance-of-plant (BOP) equipment before successful completion of a PRA for a standardized plant. It may be possible to compensate for lack of design detail by providing appropriate interface performance specifications. To conserve resources in the conduct of licensing reviews, the Commission will give priority, at the time of docketing, to standard plant applications for which a substantial portion of the NSSS and BOP design has been completed.[1]

[1] See previous Commission policy statement (43 FR 38954, August 31, 1978) with respect to antitrust aspects involved in the standardization approach.

Standardization policy will be more effective in achieving its objectives when coupled with regulatory reform initiatives for amending NRC regulations to provide for early and separate approvals of sites; for early and separate approvals of standardized nuclear power plant designs, including the balance of plant to the extent practicable; and for one-step licensing in cases using standardized whole-plant designs.

VII. FURTHER RESEARCH ON SEVERE ACCIDENTS

The Commission is conducting a research program on severe accidents. This program complements the IDCOR program of industry (see below) and it includes studies on the following:

- Probabilistic risk assessment methods, including those treating external events;
- Common-cause accident contributors;
- System interactions, including analysis of systems transients involving core damage;
- Accident management, including guidelines for recovery from a core-damaging event;
- Phenomenological research on fuel and fission product behavior of damaged cores and containment response to severe loadings;
- Human factors;
- Applications research on behavior of existing systems and components in the severe accident environment;
- Fission product release and transport; and
- Safety-cost tradeoff analysis of changes in hardware.

Among other things, the research is intended to reduce the substantial uncertainty in the risk calculations that would be used in implementing our safety policy. A basic problem to be addressed by the research program is that the uncertainty in the "front end" of PRAs (the likelihood of various accident sequences) is currently believed to be optimistically biased due to (a) possible lack of completeness in identifying all possible scenarios and describing their event sequences and (b) difficulties in identifying and modeling common-cause failures. However, the "back end" (consequence estimation) of risk assessment, is currently believed to be conservatively biased because of two basic assumptions. First, the partial failure of core cooling is usually assumed to result in total core melt. Second, recent research (see NUREG-0772) indicates that radioactive releases in dominant accident sequences are likely to be substantially lower than predictions based on the conservative assumptions in current licensing requirements or the assumptions in WASH-1400. These conservatisms concern the plateout of fission products in the primary system and the fallout of airborne radionuclides inside containments. Although these biases tend to offset one another in the overall risk estimate, they interfere with the usefulness of PRA for weighing the relative merits of different design or operating features. Research is needed to reduce the interference.

Our research program on severe accidents has two distinct phases. The first phase (scheduled for completion in early 1984) is designed to answer the necessary technical questions before final regulatory decisions on severe accidents ~~targeted for early 1984~~ are made. The objectives of Phase I (see Draft NUREG-0900, "Nuclear Plant Severe Accident Research Plan") are provide the following:

- An improved methodology for probabilistic risk assessment plus a significant extension of the data base for severe accident assessment;
- Data for a better estimate of the radiological source term used to assess accident consequences; and
- A technical basis for regulatory decisions to add or modify principal design features and operating guides and procedures of existing plants with respect to their ability to prevent and mitigate severe accidents.

In Phase II (to be completed by the end of calendar year 1985), the objectives of the program are to complete development of the data base, to further improve PRA methodology and its applications, and to confirm and render more precise the bases for regulatory decisions and guidance. Of particular importance to rulemaking on standard plant designs for future applications is that the first phase of severe accident research will enable a more precise appraisal of specific design and operational refinements, especially from the standpoint of cost-effectiveness or risk/net benefit criteria. This will serve to indicate whether further reduction of risk is justifiable. In addition, better understanding of the dominant severe accident sequences and of the magnitude of radioactive releases in the first phase of the research is expected to lead to substantial improvements in emergency preparedness and procedures.

The Commission also notes a substantial commitment of industry resources for severe accident evaluation under the Industry Degraded Core Rulemaking (IDCOR) Program (see "IDCOR Program Plan," November 1981, Technology for

Energy Corporation, Knoxville, Tennessee)., ~~The IDCOR Program~~ (to be completed by mid-1983). ~~is designed to develop in an expeditious manner, within the constraints of the best available data and methods, a comprehensive, integrated, technically sound position to help support the the NRC severe accident decision process.~~ In support of the IDCOR Program, the Nuclear Power Division of the Electric Power Research Institute has scheduled a number of ~~important~~ projects under its Degraded System Technology Program. The Commission feels it is prerequisite to the objectives and schedules set forth in this Policy Statement on Severe Accidents that the IDCOR Program continue on its present course and schedule.

We do not expect our present views on severe accident considerations to change substantially as a result of ongoing NRC-sponsored or industry research with respect to the fundamentals of the present designs and their general adherence to our safety policy. However, it is possible--though not necessarily likely for any or all classes of nuclear power plants reactors--that new information will demonstrate the desirability of certain lesser changes such as improved reliability of some engineered safety features and addition of filtered vents to some types of containment and design features that would reduce the risk of sabotage. and earthquakes. Also, we expect research results to permit further risk reduction by identifying worthwhile refinements in the design of operating reactors plants or their operating practices rather than indicating major redesign needs. The research will also help to develop more accurate probabilistic risk assessment methods for use in regulatory decision-making and to provide greater assurance of adequate protection of public health and safety.

VIII. TREATMENT OF SEVERE ACCIDENTS IN ONGOING LICENSING PROCEEDINGS

The Commission has considered the question of whether additional regulations should be issued at this time to require more capability to mitigate the consequences of severe accidents in operating plants and plants under construction. Although, as noted above, there are large programs presently ongoing that will provide information related to this question, they have not yet produced significant new insights into consequence mitigation features sufficient to support further regulatory changes, nor have they yet shown a clear need to add such features.

There are presently two rules, one final and one proposed, on hydrogen control and related matters (46 FR 58484, December 2, 1981, and 46 FR 62281, December 23, 1981). These rules are intended to provide reasonable assurance, pending generic resolution, that the risk of degraded-core accidents for plants designed in accordance with current regulatory requirements is acceptable. Accordingly, individual licensing proceedings are not appropriate forums for a broad examination of the Commission's regulatory requirements relating to control and mitigation of accidents more severe than the design basis. Similarly, notwithstanding the Class 9 accidents review requirements for environmental hearings of the Commission's Statement of Interim Policy on "Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969" (45 FR 40101, June 13, 1980), the capability of current designs or procedures (or alternatives thereto) to control or mitigate severe accidents should not be addressed in case-related safety hearings. Likewise, our new rule for pending construction permits (47 FR 2286, January 15, 1982) is sufficient for licensing of that class of plant insofar as severe accident management or consequence mitigation is concerned.

The ongoing programs of severe accident study and research described in Section VII of this Policy Statement will provide new information. ~~about severe accidents.~~ The Commission will ensure that these programs are closely coordinated and will concentrate on specific analyses and experiments needed for operating plants and plants under construction and on new standardized designs for future construction permit applications. The research will be designed to furnish information for regulatory decisions regarding features for accident prevention and management as well as consequence mitigation. The research will also improve our understanding of plant response to severe accidents so that their characteristics can be implanted into operator training and procedures. The intent is to obtain sufficient information in about two years to complete policy development and decisionmaking on severe accidents for all classes of plants. Confirmatory research may extend another several years.

In this regard, the Commission notes that much of the work to be done by the staff and its contractors as part of the Severe Accident Research Plan can be applied to light-water reactors either yet to be designed or to reactors now in operation. In some cases, the value of a change may be realized on either old or new designs. Examples of this would be changes in operator training or procedures for severe accidents and the addition of hydrogen ignition systems. In other cases, the cost of design variations could only be justified for new designs. Examples would be variations in the construction material of the containment basemat or variations in the ultimate strength of containment (see Section IX of this Policy Statement).

As described in Section VII, information on the potential for degraded-core accidents (to the extent allowed by the existing data base) is being assembled and assessed under the Industry Degraded Core Rule-making Program. This effort is directed solely towards the current generation of operating plants. Thus, there are concurrent interests of NRC and IDCOR in assessing costs and relative benefits of potential changes in design and operations for operating plants. Accordingly, we expect that our staff will meet periodically with IDCOR staff to review progress, assess the safety significance of new information, and ensure that, to the extent feasible, the programs of study are closely coordinated and complementary.

Moreover, we expect the staff and the industry to interact periodically with the ACRS on severe accident research matters applicable to plants in operation or under construction or applicable to standard designs under review for construction permit applications. The staff should exchange views initially with the ACRS to agree on a tabulation of the fundamental severe accident questions and on the approach to answering these questions for the various classes of plants operating or planned. As the programs progress in NRC and industry, the ACRS ~~should~~ will be asked to review progress and offer suggestions for change where needed to answer these fundamental questions.

The Commission will conduct an annual review of severe accident research to determine progress and to ascertain whether any substantial and significant new information has been developed that would require additional rules for severe accident protection features for operating plants and plants under construction. The Commission expects to conduct this annual review twice (the first time in the Spring of 1983 and the second, one year later), finally resolving this matter for operating plants and plants under construction by mid-1984.

IX. PRESENT VIEWS ON OTHER SAFETY ISSUES AND EFFORTS IN PROGRESS

A. Striking a Balance Between Accident Prevention and Consequence Mitigation

The general thrust of Item II.B.8 of the TMI Action Plan (NUREG-0660) was for NRC and the nuclear industry to give further consideration to severe accidents beyond the design basis and, more specifically, to explore means to decrease the probability as well as mitigate the consequences of such accidents. By using this approach the Commission seeks to strengthen its defense-in-depth policy by striking a new balance between accident prevention and consequence mitigation in controlling the risk of nuclear power plant accidents.

Preventive measures to reduce the probability of severe accid[illegible] have been at the heart of reactor safety regulation for many years -- for example, most of the General Design Criteria of 10 CFR Part 50, Appendix A. Since the accident at TMI, there has been increased recognition that one of the most important systems in providing assurance of core-melt prevention following transients is a reliable decay heat removal system (DHRS). This had led NRC to approve shutdown decay heat removal requirements as an unresolved safety issue (Task A-45).

The objective of this task is to develop a comprehensive and consistent set of shutdown cooling requirements for existing and future LWRs, including the study of alternative means of shutdown decay heat removal and of diverse "dedicated" systems for this purpose. This effort is supported by numerous research tasks. It is our hope that this will result in establishing technical performance criteria for decay heat removal systems.

However, the General Design Criteria also require that there be containment systems to control the effects of severe accidents. This is a form of consequence mitigation. Consideration of specific consequence mitigative measures for the dominant core-melt accident sequences is a new ingredient in reactor safety, and further clarification of the current policy and direction of Commission thinking on this subject for new CPs is provided below.

B. Containment Strength

In exploring the need for additional design or operational features in the next generation of plants to mitigate the consequences of core-melt accidents, the Commission will emphasize actions that improve understanding of containment building failure characteristics and design features or emergency actions that decrease the likelihood of containment building failures. The Commission has learned in its licensing activities and studies since the accident at TMI that some containments are better than others for mitigating core-melt consequences. Although not specifically designed to accommodate the hostile environments resulting from severe accidents, they can contain a large fraction of the radiological inventory from a spectrum of severe accidents. For example, large, dry containments may be sufficiently capable of mitigating the consequences of a wide spectrum of core-melt accidents; hence, further requirements may be unnecessary or, at most, upgrading current requirements to gain limited improvements of their existing capability may be necessary. The Commission expects that these matters will continue to be subjects for study (e.g., in the NRC research program and in further plant-specific studies such as the Zion and Indian Point probabilistic risk assessments).

Through an integrated systems analysis it may be possible to demonstrate that other containment types exhibit a functional containment capability equivalent to that of large, dry containments. Although containment strength is an important feature to be considered in such an analysis, credits should also be given to the inherent energy and radionuclide absorption capabilities of the various designs as well as other design features that limit or control combustible gases.

A major difficulty in assessing systems behavior under the transient conditions of a core melt is the state of knowledge of the performance of containment and other consequence mitigation technologies. P -bilistic risk assessment appears to be fairly well developed for large, dry containment types, and the staff expects that comparable knowledge will be available soon for other containment types (Ice-Condenser and Mark I, II, and III).

It is clear that core-melt accident evaluations and containment failure evaluations should continue to be performed in the context of probabilistic risk assessments for a representative sample of operating plants and plants under construction and for all future plant designs. These studies should improve our understanding of the containment loading and failure characteristics for the various classes of facilities. The analyses should be as realistic as possible and should include, where appropriate, dynamic and static loadings 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 of aerosols on engineered safety features. Following the outcome of severe accident research, a decision will be made whether to establish performance criteria for containment systems.

In addition to energy absorption capabilities mentioned above, several features that may decrease the chances of containment failure for some accidents in some containment designs are listed in Item II.B.8 of the TMI Action Plan, namely:

- Filtered venting of containment;
- Core-retention devices; and
- Hydrogen control features.

The NRC has been studying these and other mitigation features and is now in a position to give the following preliminary guidance about them for the designers of plants for *new construction* permit applications.

C. Filtered-Vented Containment Systems

In future CP applications for both pressurized water reactors (PWRs) and boiling water reactors (BWRs), filtered-vented containment systems, or a variation of such systems, should be provided if these yield a cost-effective reduction in risk. Some recent information indicates these systems may not be cost-effective for large, dry containments while other studies indicate these may be of value for some pressure suppression containments such as the MK III design of General Electric, or if the risk is dominated by large seismic events. GE has also considered a wet-well vent for its standardized Mark III design. These preliminary conclusions need to be addressed and final conclusions reached for new designs before they are applied to future plants.

D. Core-Retention Devices

Over the past several years, studies (such as NUREG-0850) of large, dry containment buildings indicate that classical core-retention devices are probably not cost-effective in reducing the release of radioactive materials to the atmosphere. Post-accident flooding of the reactor cavity may be all that is necessary to establish a coolable debris bed and prevent basemat penetration. However, unique basemat designs and unique or undesirable liquid-pathway characteristics[2] should be carefully weighed in future CP applications before deciding that this concept can be dismissed. Also, the materials of construction in the basemat can reduce or eliminate aerosols, combustibles (hydrogen and carbon monoxide), and non-condensibles arising from melted core and concrete interctions. ~~These~~ Such gases and aerosols could contribute to the overpressurization threat to containment building integrity and should be considered in an integrated evaluation of the adequacy of containment performance.

[2] For example, a core-retention device is required on the floating nuclear plant because of liquid pathway issues.

E. Hydrogen Control Systems

The Commission intends to require hydrogen control systems to deal with degraded-core accidents for all dry containments, ice condenser containments, and the Mark I, II, and III containments. The requirements for plants in operation or under construction are contained in two rules (one final and one proposed) on "Interim Requirements Related to Hydrogen Control" (46 FR 58484, December 2, 1981, and 46 FR 62281, December 23, 1981). Somewhat more stringent requirements have been set forth in the CP rule. Our existing requirements for hydrogen control systems are based on a presumption that core cooling would be restored following a severe accident and that the reactor vessel and primary heat-transport systems would maintain their integrity. The cost-effectiveness of combustible gas control systems for even more severe accidents (i.e., accidents proceeding with core melt and vessel melt-through and large combustible gas releases) should be examined for future CP applications.

F. Reliable Containment Heat Removal

The staff is studying the need for more reliable subsystems for containment heat removal ~~systems~~ (in addition to systems normally provided in the past) as ~~an option~~ possible alternatives to filtered venting for prevention of gradual over-pressurization failure of the containment building. The cost-effectiveness of this alternative should be considered in the design of plants for new CP applications. In addition to a reduction of probability of gradual over-pressurization failure, the effective design of containment heat removal subsystems may also reduce the source term for the release of radioactive materials to the environment. Research tasks directed to these alternatives could provide information useful to the cost-effectiveness analysis of alternatives and possibly in establishing technical performance criteria if these subsystems prove cost-effective. Applicants for standard design approvals or construction permits should give special consideration to reliability of decay heat removal systems as a margin of conservatism to allow for the limitations of risk assessment methods for extraordinary events such as earthquakes and sabotage.

G. Other Consequence Mitigation Measures

There are other issues listed in Item II.B.8 of NUREG-0660 that the Commission believes have minimal value for improved safety and, therefore, need not be considered further: namely, effects of severe accidents at multi-unit sites and post-accident recovery plans. ~~Another~~ One item deserving consideration, however, is the location outside containment of systems that could become highly radioactive following a severe accident. This item is not a policy question, but it is a matter of good engineering practice. The Commission expects that designers and applicants for future plants will show ~~how~~ that these systems have been located to facilitate human access ~~to buildings outside containment~~ and to enhance their long term, post-accident control and maintenance. ~~of the systems.~~

In general, core-melt consequence mitigation design features and procedures should be evaluated on as realistic a basis as possible. Considering the low probability of core-melt accidents, the Commission does not intend to require the use of conservative design criteria and analysis methods of the sort that have been applied to engineered safety features (safety-related equipment) required by NRC regulations for design basis accidents. Nonetheless, it is important to note that there may be more extreme design conditions for these mitigation systems that might compromise safety-related systems. For example, if post-accident inerting is being considered for hydrogen control following a severe accident, then the inadvertent inerting of the containment building should not violate requirements appropriate for a design basis accident (i.e., service level A for steel containment buildings, including the effects of buckling). It is also important that attendant risks be taken into account in considering design and operational improvements for core-melt mitigation. Attendant risks introduced by new systems (e.g., inadvertent operation of a system for filtered venting of containment) are an important consideration.

H. External Events, Human Errors, and Sabotage

Another class of issues is the relation to severe accident considerations of sabotage and external events such as floods, winds, and earthquakes, as well as other accident initiators that are difficult to quantify, including multiple human errors and design errors. The Commission has addressed external events and human errors in its "Proposed Policy Statement on Safety Goals for Nuclear Power Plants" (47 FR 7023, February 17, 1982) and has invited public comment on the most appropriate way of treating them. Although the Commission has explicitly excluded sabotage from the safety goal policy statement, the Commission recognizes the merit of providing guidance on plant designs that inhibit sabotage.

Pending decision on the final content of the safety goal policy statement, the Commission expects that applicants for standard design approvals will address these issues in their Safety Analysis Reports. Along with external events and human errors, these reviews will include design considerations to inhibit sabotage. Special attention should be paid to the potentially conflicting design objectives that may arise from safety and sabotage considerations. Applicants for standard design approvals or construction permits are to give specific consideration of plant design features that would decrease the probability of damage from sabotage.

In addressing potential accident initiators (including earthquakes, sabotage, and multiple human errors) where empirical data are limited and residual uncertainty is large, the use of conceptual modeling and scenario assumptions in Safety Analysis Reports will be helpful. They should be based on the best qualified judgments of experts, either in the form of subjective numerical probability estimates or qualitative assessments of initiating events and causal linkages in accident sequences. In addition to this design analysis approach for new plants, the Commission's continuing practice of conservatism and use of the defense-in-depth concept for the design basis required by current regulations are intended to provide the requisite margin of protection for accident initiators of these kinds.

I. Siting Policy

Appropriate site selection can hold significant implications for reducing the contribution to overall risk of severe nuclear accidents from external event initiators such as earthquakes, floods, and tornadoes. Moreover, site characteristics such as meteorology and terrain have significant influence on the distribution and dispersal of any accidental releases of radioactive materials. Also, the population distribution in the vicinity of the site affects the magnitude and location of potential consequences from radiation releases.

Current siting regulations are set forth in 10 CFR Part 100, and siting guidance is given in Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations." Siting policy and planning guidance for further improvements are set forth in "U. S. Nuclear Regulatory Commission Policy and Planning Guidance, 1982" (NUREG-0885, Issue 1, January 1982, page 10) as follows:

Policy

> Siting criteria for nuclear power plants and other major nuclear facilities need improvement. The staff has been working to prepare in the very near term modified regulations concerning the siting of nuclear power plants. The Commission has decided to better define its safety objectives and better characterize the radioactive source term before proceeding with new siting regulations.

Any new siting rule will be consistent with new radioactive source term information for severe accidents that ~~will~~ are expected to be available in mid-1983. A program plan for issuing a siting rule that is consistent with the Commission's reassessment of the source term for radioactive material releases and the Commission's future policy on safety goals will be developed following completion of these actions. Based on staff work to date, the new siting rule is expected to apply to future sites only and to be a refinement of present siting guidance rather than a drastic revision of it.

X. IMPLEMENTATION GUIDELINES FOR SEVERE ACCIDENT POLICY

Pending final resolution of current NRC initiatives regarding legislative and administrative regulatory reforms, as well as safety goals and their implementation plans, the Commission sets the following conditions for standard designs for reference in future CP applications or in any reactivations of previously docketed CP applications:

(1) Demonstration of compliance with current Commission regulations,

(2) Completion of a Probabilistic Risk Assessment before standard design approval through rulemaking. The applicant will be required to install those design features for prevention, management, or mitigation of severe accidents that are considered in light of Section IX above and shown to be cost-effective in the course of the rulemaking for standard design approval;

(3) Completion of a staff review of the standard design with a conclusion of safety acceptability; the review will be based upon the updated version of the Standard Review Plan (NUREG-0800) and 10 CFR 50.34(g) that requires applicants to evaluate differences from the Standard Review Plan (see 47 FR 11651, March 18, 1982, and 47 FR 15569, April 12, 1982);

(4) Consideration of all applicable Unresolved Safety Issues; and

(5) Adherence to the requirements coming from the experience at TMI and set forth in the CP Rule 10 CFR 50.34(f) (47 FR 2286, January 15, 1982).

Regarding the last item, the CP Rule applied initially to a narrow group of CP applications. However, the Commission intends that the CP Rule become a minimum requirement for new plants and, in due course, intends to modify the regulations to that effect. For those CP applications and reactivations of previously docketed CP applications meeting the guidelines above, the Commission expects that no additional fundamental design requirements relating to severe accidents will be issued, unless new safety information shows an unacceptably wide departure from the safety goals and numerical guidelines that may be issued by the Commission.

The only exception to the conditions listed above is that, between now and completion of a review of a standard plant design for severe accident considerations, the Commission will grant CPs, under its previous standardization policy, to applicants for plants referencing a standard design approval supplemented by a showing of conformance to the CP Rule, under the conditions cited in Section VI, "Standardization Policy," above. Regulatory decisions affecting operating plants and plants under construction, regarding any safety requirements being imposed on plants of new design, will be made only after due consideration of the safety-cost tradeoff criteria and available new research information on severe accidents.

LIST OF SUBJECTS IN 10 CFR PART 50

Antitrust, Classified information, Fire prevention, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, and Reporting requirements.

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Dated at Washington, D. C., this day of , 1982.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,
Secretary of the Commission.