Docket File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 16, 1994

Docket Nos. 50-387
and 50-388

See Reports

Mr. David A. Lochbaum
80 Tuttle Road
Watchung, New Jersey 07060

Mr. Donald C. Prevatte
7924 Woodsbluff Run
Fogelsville, Pennsylvania 18051

Gentlemen:

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2, SPENT FUEL POOL COOLING ISSUE (TAC NO. M85337)

We received your letter dated January 24, 1994 and appreciate your continued comments on the technical issues involved in the loss of spent fuel pool cooling scenarios raised in your November 27, 1992, 10 CFR Part 21 report and subsequent correspondence. As you know from our conversation on February 10, 1994 and from our recent transmittals of various technical documents, our technical review is continuing as described in the November 15, 1993 action plan.

As discussed in the action plan, the staff is examining in detail, the technical issues raised in your Part 21 report. However, the action plan does not discuss the overall regulatory process governing the review of the spent fuel pool (SFP) cooling issue, or indeed, any other technical issue raised regarding existing plants that may be brought to the staff's attention. The staff felt it was appropriate to present a discussion of regulatory issues and licensing issues as a response to your January 24, 1994 letter, rather than wait to discuss them in the planned safety evaluation.

Enclosure 1 to this letter discusses the process used by the staff to review technical issues with potential safety significance that arise after the plant licensing process is completed and an operating license is issued. The process is used by the staff in order to implement specific Commission policy on the treatment of safety issues raised regarding existing facilities. One key element in determining the nature of the review of any particular issue brought to light after plant licensing is the licensing basis of the facility. While plant licensing reviews, were, and still are, conducted according to the existing staff technical guidelines in existence at the time of the particular licensing review, the licensing basis for any facility is unique. The staff has reviewed the unique licensing basis of the Susquehanna facility, as it pertains to the issues raised in the Part 21 report, and has drawn the following conclusions:

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1. The offsite dose consequences for a boiling SFP event, considering a seismic event as a causal factor, but not considering a reactor accident

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as a causal or consequent event, were analyzed by the licensee and reviewed by the staff prior to issuance of the SSES Safety Evaluation Report (SER) NUREG-0776, "Safety Evaluation Report Related to the Operation of Susquehanna Steam Electric Station, Units 1 and 2." The SER review is silent with respect to the effect or analysis of a loss of coolant accident or other design basis event on the ability to meet the "postulated accident" requirements of General Design Criteria (GDC) 61.

2. Pursuant to 10 CFR 50.109, modification of the design approval for a facility which results from the imposition of a regulatory staff position that is new or different from a previously applicable staff position constitutes a backfit. NUREG-1409, "Backfitting Guidelines," provides guidance on implementation of 10 CFR 50.109 and amplifies the term "applicable staff position" to include positions taken by the staff in issuing the plant license.

3. The operating license SER for Susquehanna stated that the SFP cooling system complied with the guidance of Regulatory Guide (RG) 1.13 and met the requirements of GDC 61.

4. Therefore, the link between loss of SFP cooling events and design basis loss of coolant accidents (LOCA) and/or loss-of-offsite power (LOOP) events postulated by the authors of the Part 21 report cannot be considered within the original licensing basis of SSES.

5. Similarly, the operating license SER noted that the offsite dose consequences of a boiling SFP following a seismic event were below the guideline values of 10 CFR Part 100 and the 1.5 Rem thyroid guideline of RG 1.29. Nevertheless, in the SER, the staff specifically linked the acceptability of the nonseismic Category I SFP cooling and cleanup system to the existence of a seismic Category I standby gas treatment system (SGTS) that met the recommendations of RG 1.52.

6. Therefore, the ability of the SGTS to ventilate the fuel handling area during a boiling SFP event following a seismic event is considered within the existing licensing basis of the facility.

A detailed description of the licensing basis review is contained in Enclosure 2.

The staff noted your suggestion that the NRC require Pennsylvania Power and Light Company to develop a justification for interim operation of the Susquehanna facility. As described above, the staff has concluded that the LOCA and/or LOOP with boiling spent fuel pool scenarios raised in your Part 21 report are beyond the licensing basis of the Susquehanna facility. The staff is currently evaluating the SFP cooling complex at Susquehanna in light of the issues you raised. The staff has determined that, while this review is taking place, there is no undue risk to the public due to the low probability of the concurrent events leading to pool boiling during a loss of coolant accident

with or without a loss of offsite power. Therefore, the staff does not feel a justification for continued operation is necessary for the issues raised in the Part 21 report.

The staff has concluded that the boiling of both spent fuel pools following a seismic event is part of the licensing basis of the Susquehanna facility. In a letter dated March 7, 1994, the staff requested that the licensee provide an evaluation of the performance of the standby gas treatment system in light of this licensing basis event. Should the licensee's review determine that the standby gas treatment system is unable to perform as specified in the licensing basis, the licensee would be required to take certain actions required by the regulations. These required actions, including development of a justification for continued operation if appropriate, are described in Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability." We have included a copy of Generic Letter 91-18 as Enclosure 3.

We want to reiterate our appreciation of your efforts to bring these issues to our attention and your continuing comments on the technical issues. If you have any questions on the staff position discussed above or comments on additional issues, please do not hesitate to contact me at 301-504-1428.

Sincerely,
/S/
Joseph W. Shea, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Discussion of Regulatory Process for Review of Potentially Safety Significant Information
2. Design and Licensing Basis for Loss of Spent Fuel Pool Cooling Events at Susquehanna Steam Electric Station
3. Generic Letter 91-18

cc w/enclosures:
Mr. Robert G. Byram
Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

DISTRIBUTION w/enclosures (*w/attachments)

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Joseph W. Shea, Project Manager
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2. Design and Licensing Basis for Loss of Spent Fuel Pool Cooling Events at Susquehanna Steam Electric Station
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cc w/enclosures:
Mr. Robert G. Byram
Senior Vice President-Nuclear
Pennsylvania Power and Light
Company
2 North Ninth Street
Allentown, Pennsylvania 18101

ENCLOSURE 1

DISCUSSION OF REGULATORY PROCESS FOR REVIEW OF POTENTIALLY SAFETY SIGNIFICANT ISSUES RAISED SUBSEQUENT TO LICENSING

1.0 INTRODUCTION

This document provides a general description of how potentially safety significant information related to existing licensed reactor facilities is reviewed and processed by the Nuclear Regulatory Commission (NRC) staff. The process is traced from its legislative foundation through NRC staff guidance documents.

2.0 INITIAL LICENSING PROCESS

Title 10 of the Code of Federal Regulations (10 CFR) contains the rules and requirements instituted by the Commission to ensure that the legislatively mandated missions of the NRC are achieved. For the existing population of power reactors, requirements pertaining to the domestic licensing of nuclear production and utilization facilities are detailed in Part 50 of the Commission's regulations (10 CFR Part 50). The regulations define the types of facilities for which a Commission license is required (10 CFR 50.10), describe the types of information required from applicants for licensing consideration (10 CFR 50.30 through 50.34) and contain provisions for issuance of a license upon determination that an application meets the standards and requirements of the Atomic Energy Act of 1954 and the Commissions regulations (10 CFR 50.57).

Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," (GDC), describes the principal design criteria that apply to those facility systems, structures and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Applicants for construction permits are required by 10 CFR 50.34 to provide information on their principal design criteria and the design bases for the proposed facility as well as information on the relationship between the principal design criteria and the design bases. Applicants for operating licenses are required by 10 CFR 50.34 to submit a final safety analysis report that includes information on the design basis of the facility and presents a safety analysis of the structures, systems and components of the systems as a whole.

In reviewing a prospective licensee's application, the staff uses various guidance documents to evaluate the information provided in the application against the Commission's requirements, including for example, the GDC requirements. The various staff review guidance documents have evolved over time. Currently, systematic review guidance is contained in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (SRP). The SRP contains a general step-by-step approach that the review staff employs to provide reasonable verification that the applicable safety criteria have been met.

Staff review of a particular application is completed when the staff

determines that, based on the guidance provided in the SRP and other applicable technical documents (such as Branch Technical Positions and Regulatory Guides), the proposed facility meets all applicable Commission regulations or that the applicant has provided adequate justification for relief or exemption from specific regulations. In addition to the staff review, the Atomic Energy Act requires that a public hearing be held before issuance of a construction permit for a nuclear power plant. Additional hearings, although not mandatory, were held during the operating license review process for most existing nuclear power plants. The hearing process ensures that properly raised and admitted issues and concerns related to a specific application are aired and evaluated. The licensing process also provides for review by the Advisory Committee on Reactor Safeguards (ACRS).

Based on the determination that the facility meets the applicable regulations, the Commission can make a finding that there is reasonable assurance that proposed facility can be operated without endangering the health and safety of the public. Based on the finding of reasonable assurance of no undue risk to the health and safety of the public, as well as certain other findings related to the technical and financial qualifications of the applicant and environmental considerations, the Commission may issue an operating license.

The NRC recognizes that the review process remains an evolving one. At times, new technical information may come to light that was not considered during the licensing review. For example, the 1979 accident at Three Mile Island revealed previously unconsidered weaknesses in the design of licensed reactors. After review of the technical details of the Three Mile Island experience, the Commission implemented new requirements to improve safety at power reactors. The TMI-related requirements were compiled and set forth in NUREG-0737, "Clarification of TMI Action Plan Requirements." The new requirements were applied, as appropriate, to operating facilities and facilities under construction.

3.0 SEVERE ACCIDENT POLICY STATEMENT

As the population of operating reactors matured and after the bulk of the post-TMI requirements were implemented, the Commission sought to establish a more methodical and more predictable approach to evaluating future issues that might develop regarding the continued safe operation of existing reactors. That approach is described through a series of Commission policy statements, revised regulations and staff practices.

The Commission issued a policy statement entitled "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" (Attachment 1) published in the Federal Register on August 8, 1985 (50 FR 32138). In that statement, severe accidents are defined as those in which substantial damage is done to the reactor core whether or not there are serious offsite dose consequences. With regard to existing reactors, the Commission made the following statements:

> "On the basis of currently available information, the Commission concludes that existing plants pose no undue risk to public health and safety and sees no present basis for immediate action on generic rulemaking or other regulatory changes for these plants because of severe accident risk."

and

> "... the Commission's policy for operating reactors includes the following guidance:
>
> Operating nuclear power plants require no further regulatory action to deal with severe accident issues unless significant new information arises to question whether there is adequate assurance of no undue risk to public health and safety.
>
> In the latter event, a careful assessment shall be made of the severe accident vulnerability posed by the issue and whether this vulnerability is plant or site specific or of generic importance.
>
> The most cost-effective options for reducing this vulnerability shall be identified and a decision shall be reached consistent with the cost effectiveness criteria of the Commission's backfit policy as to which option or set of options (if any) are justifiable and required to be implemented.
>
> In those instances where the technical issue goes beyond current regulatory requirements, generic rulemaking will be the preferred solution. In other cases, the issue should be disposed of through the conventional practice of issuing Bulletins and Orders or Generic Letters where modifications are justified through backfit policy..."

In other words, the Commission established that plants which had been found to meet the Commission's existing requirements posed no undue risk to public health and safety. The policy statement establishes a clear link between a determination that a plant meets existing requirements and the position that a plant poses no undue risk. The Commission specifically reiterated this point in a Staff Requirements Memo (SRM) dated June 15, 1990 (Attachment 2). In that SRM, the Commission stated:

> "... the presumption is that compliance with our regulations provides adequate protection. The converse, however, is not true, i.e. adequate protection does not necessarily require compliance with the body of our regulations."

4.0 BACKFIT PROCESS

As discussed above, the severe accident policy statement made reference to the backfit process. The Backfit Rule, 10 CFR 50.109, defines what staff actions

are considered backfits and imposes requirements on the staff for evaluation and documentation of backfits. More detailed guidance on implementation of 10 CFR 50.109 is spelled out in NUREG-1409, "Backfitting Guidelines" (Attachment 3). The backfit rule states:

> "Backfitting is defined as the modification of or addition to systems, structures, components or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission rules or the imposition of a regulatory staff position interpreting the Commission rules that is either new or different from a previously applicable staff position..."

NUREG-1409 provides further guidance on what constitutes an applicable staff position. An applicable staff position is a requirement or position already specifically imposed on or committed to by a licensee. Such positions include NRC staff positions that are documented explicit interpretations of more general regulations and are contained in documents such as the Standard Review Plan, branch technical positions, regulatory guides, generic letters and bulletins.

The baseline determination that a plant meets Commission requirements is documented in the operating license safety evaluation for a facility and is itself an "applicable staff position". That determination results from the review of the licensee's application and is performed in accordance with the staff review guidance available at the time. Should information that was not considered during the design review subsequently come to light, the staff must follow the requirements of the backfit rule as it evaluates the new information.

The Backfit Rule and NUREG-1409 describe the three situations in which the staff may pursue a backfit and impose conditions and requirements for each of those situations. In two of those situations, referred to as "compliance backfits" and "adequate protection" backfits, the staff is required to impose the backfit.

Compliance backfits are modifications determined by the staff as necessary in order for the facility to meet existing requirements or commitments (i.e. the existing licensing basis). In this case, the staff is required to prepare a documented evaluation that the modification is necessary to bring the facility into compliance with its license, with the rule and orders of the Commission or with licensee's written commitments.

The staff may also require a backfit if a modification goes beyond the existing licensing basis, but is determined to be necessary to ensure adequate protection of the public health and safety. In this case, the staff is required to prepare a documented evaluation and finding on the basis for invoking the adequate protection principle. The staff and the Commission have had extensive discussions on what constitutes adequate protection. In SECY-

89-102, "Implementation of Safety Goal Policy" (Attachment 4), the staff comments on the usefulness of establishing a more workable definition of adequate protection but makes it clear that specific quantitative tools alone are not necessarily the appropriate measure of "adequate protection." The Commission responded to SECY-89-102 in the SRM dated June 15, 1990. In that SRM, the Commission stated:

> "The Commission believes that 'adequate protection' is a case-by-case finding based on evaluating a plant and site combination and considering the body of our regulations....It is not necessary to create a generic definition of adequate protection..."

An extensive discussion of the concept of adequate protection is provided in the supplementary information accompanying the June 1988 change to the Backfit Rule (Attachment 5). Those discussions are consistent with Commission statements made in the June 15, 1990 SRM and confirm that adequate protection must be determined on a case-by-case basis and with a substantial reliance on engineering judgement.

Finally, the staff may require a backfit if a modification is deemed necessary, not for compliance or for assurance of adequate protection, but because it would provide a substantial increase in the overall protection of overall publice health and safety and is a cost-justified safety enhancement. In this case, the staff must prepare a detailed regulatory and cost-safety benefit analysis of the proposed modification.

When new information is brought to light after issuance of a particular plant license, the staff must first clearly establish and articulate the existing licensing basis of the facility. As described in previous paragraphs, the licensing basis is limited to statements made by the staff or commitments made by the licensee in licensing documents (FSAR, SER etc.). In cases where there is conflict or confusion between the licensee's FSAR and the staff's SER, the staff's SER establishes the licensing basis.

If a review of the licensing basis determines that the new information falls outside the scope of the existing licensing basis, the staff must judge whether action is necessary to ensure the adequate protection of the public health and safety. As discussed above, the fact that new information falls outside of an existing licensing basis or may not be addressed by existing regulations does not mean that adequate protection is not provided. In reviewing issues that fall outside of an existing licensing basis or that are not addressed by existing regulations, the staff considers all related information in evaluating protection of public health and safety, including probability and consequences of related events. During such a review, the staff must consider the ability of all existing facility systems, structures and components, operating under expected or realistic conditions, to provide continued protection of the public health and safety.

If no action is required for continued assurance of adequate protection of the public health and safety, the staff can pursue the regulatory initiative to

determine if any cost-justified safety enhancement which would provide a substantial increase in protection of the public health and safety is possible. General guidance on safety benefits is given in the Commission's Safety Goal Policy Statement (Attachment 6). In the Safety Goal Policy Statement , the Commission adopted qualitative safety goals, supported by quantitative health effect objectives for use in the regulatory decision making process. The staff and Commission have worked since the issuance of that policy statement to develop an appropriate and workable implementation process for those goals. The staff's and Commission's discussions are presented in a number of documents including SECY-89-102, "Implementation of Safety Goal Policy," the SRM dated June 15, 1990, SECY-91-270, "Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy," (Attachment 7) and an SRM dated February 21, 1992 (Attachment 8). One cited measure of safety benefit is a postulated reduction in core damage frequency (CDF). SECY-91-270 provides general guidelines on the magnitude of CDF reduction appropriate for triggering further review of cost-benefit. It is clear from all of the above documents, however, that while cost benefit/ safety benefit analyses are considered on a plant specific basis, specific quantitative safety goals are not to be used in individual plant licensing decisions. Rather they are to be applied to generic regulatory initiatives.

5.0 SUMMARY

The staff seeks to evaluate any information with possible safety significance that is brought to its attention. When that information is brought to light subsequent to the licensing of a particular facility, the staff must conduct its review within the context of the backfit rule. While the backfit rule does not restrict the scope and depth of the staff's review for any one issue, it imposes requirements on the staff for implementing any initiatives that may develop from the review of that information. For reviews that fall outside of the existing licensing basis for a particular facility, the staff must determine if adequate protection of the public health and safety is still assured. In making that determination, the staff must use a variety of quantitative and qualitative tools at its disposal. Such tools can include probabilistic as well as deterministic models, consideration of existing safety and non-safety systems and consideration of operator action to mitigate the potential safety consequences contained in the new information.

Attachments

1. Policy Statement, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," published in the Federal Register on August 8, 1985 (50 FR 32138)

2. Staff Requirements Memo, dated June 15, 1990, Subject: SECY-89-102- Implementation of the Safety Goals

3. NUREG-1409, "Backfitting Guidelines," dated July 1990

Attachments (cont'd)

4. SECY Paper, SECY-89-102, "Implementation of Safety Goal Policy," dated March 30, 1989

5. Final Rule, "Revision of Backfitting Process for Power Reactors," dated May 31, 1988

6. Policy Statement; Correction and Republication, "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement; Correction and Republication," published in the Federal Register on August 21, 1986 (51 FR 30028)

7. SECY Paper, SECY-91-270, "Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy," dated August 27, 1991

8. Staff Requirements Memo, dated February 21, 1992, Subject: Briefing on Status of Safety Goal Policy Statement (SECY-91-270), 10:00 A.M., Friday, January 17, 1992, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to Public Attendance)

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants

AGENCY: Nuclear Regulatory Commission.

ACTION: Policy statement.

SUMMARY: This statement describes the policy the Commission intends to use to resolve safety issues related to reactor accidents more severe than design basis accidents. Its main focus is on the criteria and procedures the Commission intends to use to certify new designs for nuclear power plants. This policy statement is a revision of the "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation" that was published for comment on April 13, 1983 (48 FR 16014). An advance notice of proposed rulemaking, "Severe Accident Design Criteria," published on October 2, 1980 (45 FR 65474) is being withdrawn by a notice published elsewhere in this issue.

FOR FURTHER INFORMATION CONTACT: Miller B. Spangler, Special Assistant for Policy Development, Division of Systems Integration, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington D.C. 20555, Telephone: (301) 492-7305.

SUPPLEMENTARY INFORMATION: This policy statement sets forth the Commission's intentions for rulemakings and other regulatory actions for resolving safety issues related to reactor accidents more severe than design basis accidents. The main focus of this statement is on decision procedures involving staff approval or, optionally, Commission certification of new standard designs for nuclear power plants. It also provides guidance on decision and analytical procedures for the resolution of severe accident issues for other classes of future plants and for existing plants (operating reactors and plants under construction for which an operating license has been applied). Severe nuclear accidents are those in which substantial damage is done to the reactor core whether or not there are serious offsite consequences. On October 2, 1980, the Commission issued an advance notice of proposed rulemaking, "Severe Accident Design Criteria," that invited public comment on long-term proposals for treating severe accident issues (45 FR 65474). By another notice published elsewhere in this issue the Commission is withdrawing this advance notice of proposed rulemaking.

This policy statement is a revision of the "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation" published for public comment on April 13, 1983 (48 FR 16014). Twenty-six letters of comment on the proposed policy statement were received. The nuclear industry generally supported the proposed policy statement and suggested several modifications. Much of the criticism of the proposed policy statement by environmental groups and other interested persons focused on a perception of over-reliance on probabilistic risk assessment, especially when coupled with the Commission's "Safety Goal Development Program" (48 FR 10772, March 14, 1983). The Policy Statement was revised as a result of these suggestions and criticisms as well as comments by the Advisory Committee on Reactor Safeguards.

Many changes have already been implemented in existing plants as a result of the TMI Action Plan (NUREG-0660 and NUREG-0737).[1] Information resulting from NRC- and industry-sponsored research, and data arising from construction and operating experience. On the basis of currently available information, the Commission concludes that existing plants pose no undue risk to public health and safety and sees no present basis for immediate action on generic rulemaking or other regulatory changes for these plants because of severe accident risk. The Commission has ongoing nuclear safety programs that include: the resolution of new and several other Unresolved Safety Issues and Generic Safety Issues; the Severe Accident Source Term Program; the Severe Accident Research Program; operating experience and data evaluation regarding failure of certain Engineered Safety Features and safety-related equipment, human errors, and other sources of abnormal events; and scrutiny by the Office of Inspection and Enforcement to monitor the quality of plant construction, operation, and maintenance. Should significant new safety information become available, from whatever source, to question the conclusion of "no undue risk," then the technical issues thus identified would be resolved by the NRC under its backfit policy and other existing procedures, including the possibility of generic rulemaking where this is justifiable.

One important source of new information is the experience of NRC and the nuclear industry with plant-specific probabilistic risk assessments. Each of these analyses, which provide a detailed assessment of possible accident scenarios, has exposed relatively unique vulnerabilities to severe accidents. Generally, the undesirable risk from these unique features has been reduced to an acceptable level by low-cost changes in procedures or minor design modifications. Accordingly, when NRC and industry interactions on severe accident issues have progressed sufficiently to define the methods of analysis, the Commission plans to formulate an integrated systematic approach to an examination of each nuclear power plant now operating or under construction for possibly significant risk contributors that might be plant specific and might be missed absent a systematic search. Following the development of such an approach, an analysis will be made of any plant that has not yet undergone an appropriate examination and cost-effective changes will be made, if needed, to ensure that there is no undue risk to public health and safety. In implementing such a systematic approach, plants under construction that have not yet received an Operating License will be treated essentially the same as the manner by which operating reactors are dealt with. That is to say, a plant-specific review of severe accident vulnerabilities using this approach is not considered to be necessary to determine adequate safety or compliance with NRC safety regulations under the Atomic Energy Act, or to be a necessary or routine part of an Operating License review for this class of plants.

Regarding the decision process for certifying a new standard plant design—an approach the Commission strongly encourages for future plants—the Policy Statement affirms the Commission's belief that a new design for a nuclear power plant can be shown to be acceptable for severe accident concerns if it meets the following criteria and procedural requirements:

- Demonstration of compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island requirements for new plants as reflected in the CP Rule [10 CFR 50.34(f), 47 FR 2286];
- Demonstration of technical resolution of all applicable Unresolved Safety Issues and the medium- and high-priority Generic Safety Issues, including a special focus on assuring the reliability of decay heat removal

[1] Documents referenced in this Policy Statement are available for inspection at the NRC's Public Document Room, 1717 H Street, NW, Washington, D.C.

systems and the reliability of both AC and DC electrical supply systems;

- Completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety; and

- Completion of a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by PRA.

Custom designs that are variations of the present generation of LWRs will be reviewed in future construction permit applications under the guidelines identified for approval or certification of standard plant designs.

Because this policy statement is just one part of a larger program, including the Severe Accident Research Program, for resolving severe accident issues, the NRC staff is publishing concurrently with this Policy Statement a report on "NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulation" (NUREG-1070). In this report the Policy Statement is reprinted along with other information and appendices that provide perspective on the development and implementation of this policy and how it relates to other features of the Severe Accident Program. A copy of NUREG-1070 will be available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C. Copies of NUREG-1070 may be purchased by calling (202) 275-2060 or (202) 275-2171 or by writing to the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, D.C. 20013-7082 or the National Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161.

Policy Statement

A. Introduction

The focus on severe accident issues in this Policy Statement is prompted by the staff's judgment that accidents of this class, which are beyond the substantial coverage of design basis events, constitute the major risk to the public associated with radioactive releases from nuclear power plant accidents. A fundamental objective of the Commission's severe accident policy is that the Commission intends to take all reasonable steps to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur.

On April 13, 1983, the U.S. Nuclear Regulatory Commission issued for public comment a "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation" (48 FR 16014). The public comments have been reviewed, and, on the basis of further study and consultation, the Commission is issuing the present Policy Statement as a guide to regulatory decision making on the treatment of severe accident issues for existing and future nuclear reactors[2] with special focus on procedures for staff approval or, optionally, Commission certification of new standard plant designs.[3]

In line with its legislative mandate to ensure that nuclear power plants should pose no undue risk to public health and safety, the Commission has examined an extensive range of technical issues relating to severe accident risk that have been identified since the accident at Three Mile Island. Following implementation of numerous modifications of plant design and regulatory procedures as developed through the TMI Action Plan (NUREG-0660 and NUREG-0737) and other Commission deliberations, the Commission concludes (based on current information and analyses) that existing plants do not pose an undue level of risk to the public. On this basis, the Commission feels there is no need for immediate action on generic rulemaking or other regulatory changes for these plants because of severe accident risk. However, the occurrence of a severe accident is more likely at some plants than at others. At each plant there will be systems, components or procedures that are the most significant contributors to severe accident risk. The intent of this policy statement is to provide utilities with basis for development of Commission guidance that will allow identification of these contributors and development of the appropriate course of action, as needed to assure acceptable margins of safety. In all cases, the commitment of utility management to the pursuit of excellence in risk management is of critical importance. The term "risk management" includes accident prevention, accident management to curtail or retard its progression, and consequence mitigation to further limit its effects on public health and safety. The Commission plans to formulate an approach for a systematic safety examination of existing plants to determine whether particular accident vulnerabilities are present and what cost-effective changes are desirable to ensure that there is no undue risk to public health and safety. In implementing such a systematic approach, plants under construction that have not yet received an Operating License will be treated essentially the same as the manner by which operating reactors are dealt with. That is to say, a plant-specific review of severe accident vulnerabilities using this approach is not considered to be necessary to determine adequate safety or compliance with NRC safety regulations under the Atomic Energy Act, or to be a necessary or routine part of an Operating License review for this class of plants.

The main purposes of this Policy Statement follow:

- To clarify the procedures and requirements for licensing a new nuclear plant;

- To re-examine the need for the generic rulemaking proceeding contemplated in the TMI Action Plan commitment (NUREG-0660, Task II.B.8) on degraded core accidents, currently referred to as severe nuclear reactor accidents;

- To avoid unnecessary delays of plants now under construction;

- To close out for now severe accident issues for existing plants (those in operation and under construction) without imposing further backfits unless this can be justified by new safety information; and,

- To achieve improved stability and predictability of reactor regulation in a manner that would merit improved public confidence in our regulatory decision making.

The policies presented in this statement will lead to amendment of NRC regulations, standard review plans for licensing actions, or other decision procedures and criteria as part of NRC's ongoing Severe Accident Program. This Policy Statement makes allowance for such changes as the result of the development of new safety information of significance for design and operating procedures.

[2] The term "nuclear reactor" is commonly used as a synonym for a nuclear power plant which, in addition to the Nuclear Steam Supply System, includes facilities and equipment denoted as Balance-of-Plant.

[3] For forward referenceability of a new standard design, the applicant is being afforded in this Policy Statement the flexibility of choosing between a Preliminary Design Approval (PDA), a Final Design Approval (FDA), or Design Certification (DC). The design approvals (i.e., a PDA or FDA) would be issued following the completion of the staff's review and would be subject to challenge in individual licensing hearings. The Design Certification would be issued by the Commission following a rulemaking proceeding and could not be challenged in individual hearings.

In accordance with the activities, views, and policy developments discussed in this policy Statement, the Commission believes that it is possible to complete its ongoing reviews of new plant designs 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 the ongoing NRC, Industry Degraded Core Rulemaking Program (IDCOR), and vendor research and insights from the Zion, Indian Point, Limerick, and other risk analyses. The review of standard designs for future CPs provides incentive to industry to address severe accident phenomena. Indeed, since July 1983, the staff has completed the reviews and has issued Final Design Approvals (FDAs) for two standard designs (General Electric Company's BWR/6 Nuclear Island Design, GESSAR II; and Combustion Engineering Incorporated's System 80 Design, CESSAR). A severe accident review by the NRC staff of the GESSAR II design for forward referenceability is nearly complete. The review included assessment of alternative design changes for severe accident risk reduction. In addition, the staff has been involved with pretendering review of an application for Westinghouse Electric Corporation's advanced pressurized water reactor design RESAR-SP/90. In January 1984, the NRC found the RESAR-SP/90 application for a Preliminary Design Approval acceptable for docketing and in May 1984 the application was docketed. Also, work has been continuing between NRC and the Electric Power Research Institute (EPRI) on their "LWR standardized Future Plant Design Evaluation Program."

It is assumed in this Policy Statement that, over the next 10 to 15 years, utility and commercial interest in the United States will focus on advanced light water reactors that involve improvements but are essentially based on the technology that was demonstrated in the design, construction, and operation of more than 100 of these plants in the United States. This policy should not be viewed as prejudicial to more extensive changes in reactor designs that might be demonstrated during or beyond that time period. Indeed, the Commission encourages the development and commercialization of any standard designs that might realize safety benefits, such as those achieved through greater simplicity; slower dynamic response to upset conditions involving accident precursor events; passive heat removal for loss-of-coolant accidents; and other characteristics that promote more efficient construction, operation, and maintenance procedures to enhance safety, reliability, and economy.

B. Policy for New Plant Applications

1. Introduction

No new commercial nuclear reactors have been ordered in the United States since December 1978. However, the Commission has received several applications for reference design approvals that are currently under review. A reference design is one of the options in the Commission's standardization policy. When approved by the NRC staff, a reference design could be incorporated by reference in a new CP application and, ultimately, in an Operating License (OL) application. During the corresponding CP and OL reviews, the NRC staff would not duplicate that portion of its review encompassed by its reference design approval. Therefore, even in the absence of new CP applications, in order to provide guidelines for the current reference design reviews, the Commission has recognized the need to promptly establish the criteria by which new designs can be shown to be acceptable in meeting severe accident concerns. The Commission now believes that there exists an adequate basis from which to establish an appropriate set of criteria. This belief is supported by current operating reactor experience, ongoing severe accident research, and insights from a variety of risk analyses. The resultant criteria and procedural requirements are listed below.

2. Criteria and Procedural Requirements

The Commission believes that a new design for a nuclear power plant (as well as a proposed custom plant) can be shown to be acceptable for severe accident concerns if it meets the following criteria and procedural requirements:

a. Demonstration of compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island requirements for new plants as reflected in the CP Rule [10 CFR 50.34(f)];

b. Demonstration of technical resolution of all applicable Unresolved Safety Issues and the medium- and high-priority Generic Safety Issues, including a special focus on assuring the reliability of decay heat removal systems and the reliability of both AC and DC electrical supply systems;

c. Completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety; and

d. Completion of a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by PRA.

The fundamental criteria listed above apply to the staff's review of any new design. In addressing criteria (b) and (c), the applicant for approval or certification of a reference design shall consider a range of alternatives and combination of alternatives to address the unresolved and generic safety issues and to search for cost-effective reductions in the risk from severe accidents. No cost-benefit standard has currently been certified by the Commission, although one has been proposed for trial use (NUREG-0880, Rev. 1). Such a standard, if certified, could serve as a surrogate, not only for dollar costs and benefits of a decision option, but also for other adverse and beneficial effects (soft attributes) of social significance that cannot readily be quantified in commensurate units.

The following sections explain in more detail how these criteria are to be applied to the various types of reviews that the staff may encounter. It is intended that a new design would satisfy each of the fundamental criteria listed above before *final* approval or certification. It is recognized, however, that a new design can go through different stages or levels of approval before receiving this final approval or certification. For example, a reference design can obtain a Preliminary Design Approval (PDA) and then a Final Design Approval (FDA). The unique circumstances of each design review will, therefore, require flexibility in the application of the criteria listed above. In particular, the timing of the PRA requirement may differ considerably from one review to another. In addition, the licensee is required to ensure that the intent of the safety requirements is accomplished during procurement, construction and operation.

It is recognized that there are a diversity of PRA methods. These will continue to undergo evolutionary development as the results of research programs and reliability data from operating reactors become available and as innovative uses of PRA in safety decision contexts suggest better ways to achieve the benefits of these methods while guarding against their limitations or improper uses. While learning curves of these kinds will likely continue for a

decade or more, it would nevertheless be constructive to consolidate this experience at various stages of PRA development and utilization. At the present stage of development, a number of positive uses of PRAs have been demonstrated, especially in identifying: (1) Those contributors to severe accident risk that are clearly dominant and hence need to be examined for cost-effective risk reduction measures and (2) those accident sequences that are clearly insignificant risk contributors and can therefore be prudently dismissed. In-between cases are more problematic.

Accordingly, within 18 months of the publication of this severe accident statement, the staff will issue guidance on the form, purpose and role that PRAs are to play in severe accident analysis and decision making for both existing and future plant designs and what minimum criteria of adequacy PRAs should meet. From experience to date, it is evident that PRAs could serve as a highly useful tool in assessing the risk-reduction potential and cost-effectiveness of a number of imaginative design options for new plants in comparison with design features of existing plants. The PRA guidance will describe the appropriate combination of deterministic and probabilistic considerations as a basis for severe accident decisions.

The proposed Commission Policy Statement on Severe Accidents issued on April 13, 1983 recognizes the need for striking a balance between accident prevention and consequence mitigation. 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 strike a balance between accident prevention and consequence mitigation encompassing actions that improve understanding of containment building failure characteristics and design features or emergency actions that decrease the likelihood of containment building failures. Although not specifically designed to accommodate all of the hostile environments resulting from the complete spectrum of severe accidents, they can contain a large fraction of the radiological inventory from a portion of the spectrum of such 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).

Integrated systems analysis will be used to explore whether 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.

It is clear that core-melt accident evaluations and containment failure evaluations should continue to be performed 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 on aerosols on engineered safety features. A clarification of containment performance expectations will be made including a decision on whether to establish new performance criteria for containment systems and, if so, what these should be.

The Commission also recognizes the importance of such potential contributors to severe accident risk as human performance and sabotage. The issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized as special considerations in the design and in the operating procedures developed for new plants. Likewise, the effectiveness of human performance will be emphasized in design and operating procedure development. A balanced focus will be paid to the negative impact of human performance on severe accident risk as well as its potentially positive contribution to halting or limiting the consequences of severe accident progression. Design features should be emphasized that reduce the risk of early containment failure, thus providing more time for the positive contributions of operator performance in curtailing severe accident consequences. Also, design features should be given special attention that serve to decrease the role of human error in the sequence of events leading to the initiation or aggravation of core degradation. In particular, methods of analysis and associated data bases are under development by the Commission's ongoing severe accident programs that will aid the analyses and corrective actions of both negative and positive human performance contributions to severe accident risk or its alleviation.

It is noted that some of the severe accident scenarios result in insignificant probability of offsite consequences, because of containment effectiveness. In this situation, there may be no clear basis for regulatory action because there is no substantial effect on public health or safety. However, the implementation of requirements to control occupational exposure should be considered along with the relatively small effects on public health and safety for these types of severe accidents. The resolution of cost-benefit issues in severe accident decision making is part of the NRC's Safety Goal Evaluation Program.

Although in the licensing of existing plants the Commission has determined that these plants pose no undue risk to public health and safety, this should not be viewed as implying a Commission policy that safety improvements in new plant designs should not be actively sought. The Commission fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs. This expectation is based on:

- The growing volume of information from industry and government-sponsored research and operating reactor experience has improved our knowledge of specific severe accident vulnerabilities and of low-cost methods for their mitigation. Futher learning on safety vulnerabilities and innovative methods is to be expected.
- The inherent flexibility of this Policy Statement (that permits risk-risk tradeoffs in systems and sub-systems design) encourages thereby innovative ways of achieving an improved overall systems reliability at a reasonable cost.
- Public acceptance, and hence investor acceptance, of nuclear technology is dependent on demonstrable progress in safety performance, including the reduction in frequency of accident precursor events as well as a diminished controversy among experts as to the adequacy of nuclear safety technology.

• Further progress in severe accident risk reduction is a hedge against the possibility that current risk estimates ·ith their broad ranges of uncertainty ight unwittingly have been optimistically biased.

• Although the severe accident risk of an individual plant may be acceptable in terms of its direct offsite regional consequences for public health and safety, the aggregate probability (say, over a 30-year period) that one severe accident will occur in a large population of reactors holds a separate and additive significance. Such an event would yield adverse spillover consequences for innocent parties in other regions (i.e., nuclear-oriented utilities and their customers), not to mention a changed political environment for nuclear regulation itself affecting resource costs and programmatic activities.

3. Application of Criteria for Different Types of OL and CP Applications

a. *Application of Certification of Reference Designs with No Previous FDA.* In accordance with the Commission's standardization regulations and policy, a new reference design can be submitted for approval, first as a preliminary design and then as final design. Correspondingly, the staff will issue a Preliminary Design Approval and a Final Design Approval. PDA is not, however, a prerequisite an FDA. An applicant has the option .. submit FDA-level information initially and proceed directly with an FDA review. These options remain unchanged by this Policy Statement.

After a PDA application is docketed, the preliminary design can be referenced in a new CP application. The corresponding OL application would then reference the approved final design (FDA). Of course, an approved design could also be referenced in a new CP application.

The use of an approved standard design in new CP/OL applications has received considerable attention under the Commission's legislative initiatives on single-step licensing. It should be noted that a two-step review process for a standard design approval is not, in itself, inconsistent with single-step licensing. To be most effective, single-step licensing presumes the existence of a previously approved design—essentially an FDA. This design could still be approved in a two-step process as long as both steps were completed in advance of the single-step licensing application.

The use of PRA in a two-step review process also raises a number of ·estions. Of particular concern is the timing of the PRA requirement because the completion of a comprehensive and detailed PRA may not be achievable in the absence of essentially complete and final detailed design information. Therefore, to require a complete PRA at the PDA stage would not be realistic. The Commission's recent experience, however, indicates that a substantial amount of design detail that would permit meaningful, limited, quantitative risk analysis does exist at the PDA stage. Because the Commission believes that risk analysis of this type would be a useful design tool, the Commission expects that it would be completed as part of the PDA application process. A complete risk analysis would not be a prerequisite for issuance of a PDA. However, if this risk analysis is not performed in the PDA process, it will have to be provided as part of any CP application referencing the design.

If the scope of the FDA reference design application is limited to an extent that would preclude the completion of a meaningful, comprehensive PRA, the requirement for a complete PRA may be waived. However, the applicant should still perform and submit supplementary risk analysis, to the extent practical, to demonstrate the adequacy of the proposed design. If a comprehensive PRA is not submitted for an FDA, a CP/OL applicant referencing the approved design would be required to submit a plant-specific PRA. For standard design approvals of restricted scope, additional limitations beyond the PRA aspects may exist. Use of such a standard design by the license applicant may be limited by its very nature to a two-step licensing process, namely, a Construction Permit and an Operating License issued separately. This would negate some of the benefits envisioned for an approved or certified design wherein a previously approved site could be matched with it in a one-step, combined CP/OL process.

The reference design must satisfy each of the criteria stated in Section B.2 before an FDA can be issued. For forward referenceability of a new standard design, the applicant is being afforded in this Policy Statement the flexibility of choosing between a Preliminary Design Approval (PDA), a Final Design Approval (FDA), or a Design Certification (DC). The design approvals (i.e., a PDA or FDA) would be issued following the completion of the staff's review and would be subject to challenge in individual licensing hearings. The Design Certification would be issued by the Commission following a rulemaking proceeding and could not be challenged in individual hearings. CPs or OLs, based on a reference design that has not been approved through rulemaking, shall be subject to any design changes arising from the rulemaking proceeding in accordance with the Commission's backfit policy and regulations. The design certification would be issued for a longer duration than a design approval. The specific requirements and procedures for obtaining design certifications or approvals will be established in a forthcoming revision to the Commission's Standardization Policy Statement.

b. *Approval or Certification of Reference Designs Previously Granted an FDA.* In 1983, the NRC staff issued two Final Design Approvals for reference designs. These designs were permitted to be incorporated by reference in OL applications where the corresponding CP application had referenced the PDA. However, the designs were not approved for incorporation in new CP applications. The Commission now believes that these designs are suitable for use in new CP and OL applications under the conditions specified below. Any significant changes to these designs, other than those resulting from the severe accident review, will require the designs to be considered under the provisions of Section B.3.a, i.e., as new designs.

(1) Each of the two reference design applicants with existing FDAs must request that their FDAs be amended to permit their designs to be referenced in new CP and OL applications. The request must either (i) include the information needed to satisfy each of the criteria stated in Section B.2, or (ii) provide suitable interface requirements to ensure that CP and OL applications referencing the design will satisfy each of the criteria in Section B.2. Requests in either case need not include an evaluation of how the design conforms to the Standard Review Plan (10 CFR 50.34(g)).

In the first case, the staff will amend the existing FDA upon receipt of the request to permit the design to be referenced in new CP and OL applications until the severe accident review is completed. The severe accident review must be successfully completed prior to the issuance of any new CP or OL whose applications reference the design. Upon the successful completion of the severe accident review, the staff will further amend the FDA to permit the design to be referenced in new CP and OL applications for a fixed period of time, such as five years.

In the second case, the staff will amend the existing FDA upon receipt of

the request to permit the design to be referenced in new CP and OL applications for a fixed period of time, such as five years. The amended FDA will be conditioned as appropriate to ensure that new CP and OL applications referencing the design will satisfy each of the criteria in Section B.2. The severe accident review must be completed prior to the issuance of the new CP or OL.

(2) Criterion B.2.c requires the completion of a comprehensive PRA. If a comprehensive PRA cannot be completed owing the the limited scope of the design, the applicant shall perform supplementary risk analyses to the extent practical in support of the approval or rulemaking process. As noted above, the limited scope of plant design and PRA analysis would lead to a partial loss of benefits in that a two-step CP/OL licensing process would be required in lieu of a one-step process.

(3) With regard to completion of a comprehensive PRA for a reference design, the Commission recognizes that a PRA would be more meaningful if it were based on a substantial portion of the complete facility design. Therefore, if justified to the NRC staff, completion of the PRA by the FDA applicant may be waived. If a comprehensive PRA is not submitted by the FDA applicant for the FDA, a CP/OL applicant referencing the design would be required to submit a plant-specific PRA.

A reference design applicant previously granted an FDA can pursue the same options of design approval or design certification as described in the preceding section for reference designs with no previous FDA. The FDA would be issued following the completion of the staff's review and would be subject to challenge in individual licensing hearings. The Design Certification would be issued by the Commission following a rulemaking proceeding and could not be challenged in individual hearings. CPs or OLs, based on a reference design that has not been approved through rulemaking, shall be subject to any design changes arising from the rulemaking proceeding in accordance with the Commission's backfit policy and regulations. The design certification would be issued for a longer duration than a design approval. The specific requirements and procedures for obtaining design certifications or approvals will be established in a forthcoming revision to the Commission's Standardization Policy Statement.

c. A Reactivated Construction Permit Application. Because of the many complex factors involved, the criteria and procedures for regulatory treatment of reactivated Construction Permits will be a matter of separate consideration apart from this Severe Accident Policy Statement.

d. A New Custom Plant Construction Permit Application. It is the Commission's policy to encourage the use of reference designs in future CP applications. This does not, however, preclude the use of a custom design. Custom designs shall also be reviewed against the criteria identified in Section B.2. As a result of the circumstances and timing involved in the ongoing standard design review processes, the Commission expects that most, if not all, new CP applications incorporating a reference design would be based on essentially final design information. This will result in improved safety and regulatory practices, as well as reduced time to license and construct a nuclear power plant. To obtain as much of this benefit as practicable for a custom design application, the Commission will require a CP application for a custom design to include design information that is sufficiently final and complete to permit completion of an adequate plant-specific PRA. It is possible, however, that an applicant referencing an approved or certified design in lieu of a custom plant would have in prospect a significantly reduced licensing fee since staff effort would not be required—or much less would be required—for a rereview of the approved or certified design at the CP/OL stage save for those detailed changes to accommodate unique site features or other special circumstances (e.g., innovative equipment designs to meet new ASME or IEEE codes, etc.)

C. Policy for Existing Plants

1. Some General Principles of Policy Development

The Commission has licensed about 90 nuclear plants and expects to process applications to license approximately 30 additional plants. The Commission has considered at length the question of whether generic rulemaking should be undertaken or additional regulations should be issued at this time to require more capability in operating plants or plants under construction to improve severe accident prevention, consequence mitigation, or accident management that would halt or delay further core degradation.

The TMI accident led to a number of investigations of the adequacy of design features, operating procedures, and personnel of nuclear power plants to provide assurance of no undue risk regarding severe reactor accidents. The report "NRC Action Plan Developed as a Result of the TMI-2 Accident" (NUREG-0660, May 1980) describes a comprehensive and integrated plan involving many actions that serve to increase safety when implemented by operating plants and plants under construction. The Commission approved items for implementation and these are identified in a report, "Clarification of TMI Action Plan Requirements" (NUREG-0737, November 1980). The staff issued further criteria on emergency operational facilities (NUREG-0737, Rev. 1), auxiliary feedwater system improvements (derived from NUREG-0667), and instrumentation (Regulatory Guide 1.97, Revision 2).

The TMI Action Plan led to the requirements of over 6,400 separate action items for operating reactors and five Near-Term Operating Licenses. About 90 percent of the action items approved for operating reactors are now complete and the remainder are expected to be finished by the end of fiscal year 1985. There were 132 different types of action items approved in the Action Plan (an average of 90 actions per plant). Of this total, 39 involved equipment backfit items, 31 involved procedural changes, and 62 required analyses and reports. It is impractical to quantify all of the safety improvements obtained by these many changes. Nevertheless, the cumulative effect is undoubtedly a significant improvement in safety.

Other information from NRC- and industry-sponsored research along with failure data from construction and operating experience have led to changes in existing plants. Also, the NRC/AEC has sponsored 11 plant-specific PRAs and the industry has sponsored many more. The evaluation of severe accident risk by the interrelated deterministic and probabilistic methods has identified many refinements of current design and operating practice that are worthwhile, but has identified no need for fundamental (or major) changes in design.

On the basis of currently available information, the Commission concludes that existing plants pose no undue risk to public health and safety and sees no present basis for immediate action on generic rulemaking or other regulatory changes for these plants because of severe accident risk. Moreover the Commission has ongoing programs (described in NUREG-1070 and issued concurrently with this Policy Statement) that include: the resolution of Unresolved Safety Issues and other Generic Safety Issues, including a special focus on assuring the reliability of decay heat removal systems and the reliability of both AC and DC electrical supply systems; the Severe Accident Source Term Program; the Severe Accident Research Program; operating experience and data evaluation regarding equipment failure, human errors, and other sources of abnormal events; and scrutiny by the Office of Inspection and Enforcement to monitor the quality of plant construction, operation, and maintenance. The Commission will maintain its vigilance in these programs to offset the uncertainty of whether significant safety issues remain to be disclosed. Industry research and foreign reactor experience are also meaningful sources of information.

One important source of new information is the experience of NRC and the nuclear industry with plant-specific probabilistic risk assessments is that each of these analyses, which provide a more detailed assessment of possible accident scenarios, has exposed relatively unique vulnerabilities to severe accidents. Generally, the undesirable risk from these unique features has been reduced to an acceptable level by low-cost changes in procedures or minor design modifications. Accordingly, when NRC and industry interactions on severe accident issues have progressed sufficiently to define the methods of analysis, the Commission plans to formulate an integrated systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant risk contributors (sometimes called "outliers") that might be plant specific and might be missed absent a systematic search. Following the development of such an approach, an analysis will be made of any plant that has not yet undergone an appropriate examination. The examination will include specific attention to containment performance in striking a balance between accident prevention and consequence mitigation. In implementing such a systematic approach, plans under construction that have not yet received an Operating License will be treated essentially the same as the manner by which operating reactors are dealt with. That is to say, a plant-specific review of severe accident vulnerabilities using this approach is not considered to be necessary to determine adequate safety or compliance with NRC safety regulations under the Atomic Energy Act, or to be a necessary or routine part of an Operating License review for this class of plants.

Should significant new safety information develop, from whatever source, which brings into question the Commission's conclusion that existing plants pose no undue risk, then at that time the specific technical issues suggesting undue vulnerability will undergo close examination and be handled by the NRC under existing procedures for issue resolution including the possibility of generic rulemaking where this is justifiable. However, NRC's experience suggests that safety issues discovered through operating experience programs, quality assurance programs or safety analyses often pertain to *unique* characteristics of a specific plant design and, therefore, are dealt with through plant-specific modifications of relatively modest cost rather than major *generic* design changes.

The Severe Accident Research Program as well as NRC's extensive severe accident studies of certain individual plants will aid in determining the extent to which carefully analyzed reference plants can appropriately serve as surrogates for a class of similar plants as the basis for any generic conclusions. These studies will also aid in identifying the desirable scope and approach for follow-up safety studies of individual plants. Any generic changes that are identified as necessary for public health and safety will be required through rulemaking and will be consistent with the Commission's backfit policy.

2. Policy for Operating Reactors

In light of the above principles and conclusions, the Commission's policy for operating reactors includes the following guidance:

- Operating nuclear power plants require no further regulatory action to deal with severe accident issues unless significant new safety information arises to question whether there is adequate assurance of no undue risk to public health and safety.
- In the latter event, a careful assessment shall be made of the severe accident vulnerability posed by the issue and whether this vulnerability is plant or site specific or of generic importance.
- The most cost-effective options for reducing this vulnerability shall be identified and a decision shall be reached consistent with the cost-effectiveness criteria of the Commission's backfit policy as to which option or set of options (if any) are justifiable and required to be implemented.
- In those instances where the technical issue goes beyond current regulatory requirements, generic rulemaking will be the preferred solution. In other cases, the issue should be disposed of through the conventional practice of issuing Bulletins and Orders or Generic Letters where modifications are justified through backfit policy, or through plant-specific decision making along the lines of the Integrated Safety Assessment Program (ISAP) conception.[4]
- Recognizing that plant-specific PRAs have yielded valuable insight to unique plant vulnerabilities to severe accidents leading to low-cost modifications, licensees of each operating reactor will be expected to perform a limited-scope, accident safety analysis designed to discover instances (i.e., outliers) of particular vulnerability to core melt or to unusually poor containment performance, given core-melt accidents. These plant-specific studies will serve to verify that conclusions developed from intensive severe accident safety analyses of reference or surrogate plants can be applied to each of the individual operating plants. During the next two years, the Commission will formulate a systematic approach, including the development of guidelines and procedural criteria, with an expectation that such an approach will be implemented by licensees of the remaining operating reactors not yet systematically analyzed in an equivalent or superior manner.

3. Policy for Operating License Applications for Plants Currently Under Construction

The same severe accident policy guidance applies to applications for operating licenses (OLs) as stated above for operating nuclear power plants along with the following additional item. (This item also applies to any hearing proceedings that might arise for an operating reactor.)

- Individual licensing proceedings are not appropriate forums for a broad examination of the Commission's regulatory policies relating to evaluation, control and mitigation of accidents more severe than the design basis (Class 9). The Commission has announced a policy regarding Class 9 environmental reviews and hearings in its Statement of Interim Policy on "Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969" (45 FR 40101, June 13, 1980), and expects to continue this policy. The environmental issues deal essentially with the estimation and description of the risk of

[4] See "Integrated Safety Assessment Program (ISAP)," SECY 84-133, March 23, 1984.

severe accidents. The Commission believes that considerations which go beyond that to the possible need for safety measures to control or mitigate severe accidents in addition to those required for conformance with the Commission's safety regulations or conformance with the Clarification of TMI Action Plan Requirements,[4] should not be addressed in case-related safety hearings.

The Separate Remarks of Chairman Palladino and the Dissenting Views of Commissioner Asselstine are attached.

Dated at Washington, D.C., this 30th day of July 1985.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

Separate Remarks by Chairman Palladino

I believe the Commission is on the right course with this decision. The severe accident policy statement presented here is based on the arguments contained within it, the additional support of more detailed analysis in its companion document NUREG–1070, the massive support of the many other related works of this agency and others in this field, and a logical consistency with other actions of the Commission.

In simple terms, this policy statement says that existing plants pose no undue risk to public health and safety, and that there is no present basis for regulatory changes for these plants due to severe accident risk. This conclusion on reactor safety does not lead us to dismantle our regulatory program; rather we are maintaining a vigorous program of surveillance, analysis, and evaluation to foresee possible causes of accidents and prevent them. In this perspective, the Commission has ongoing nuclear safety programs that include: unresolved safety issues; severe accident, source term and research programs; operating experience and data evaluation, and the scrutiny of plant construction, operation and maintenance. Should significant new safety information become available, from whatever source, to question the conclusion of no undue risk, then the technical issues thus identified would be resolved by the NRC under its backfit policy or other existing procedures.

The level of risk found to be acceptable is well documented in the basic works of the agency on these related subjects. The calculated frequency of severe core damage, whether mean or median value, is on the order of 1 chance in 10,000 per reactor year. For most plants, only a fraction of the calculated severe core damage sequences are likely to progress to large scale core melt. Until now, few analysts have even tried to take that fraction into separate consideration, preferring even to refer to the previously calculated value as the core melt frequency. Of the core melt sequences, typically only 1 in 10, or less, are expected to yield large releases of radioactive material. On virtually every reactor site in the United States conditions are such that, even with a large release, there is only 1 chance in 10 of any early fatality—and so on. Thus, the wealth of risk estimates before us indicate that the risk is quite low.

It is often said that one should beware of too much trust in the point estimates of probabilistic risk assessments, that one should consider the uncertainties. This we do. But some then go on to demand exact quantitative definitions of the uncertainty. This demand is a form of bottom line fallacy.

Precise statements of uncertainty come only with large amounts of data. At the very low levels of risk with which we are dealing, the occurrence of actual events is, thankfully, very rare indeed. Thus, we cannot have exact quantitative estimates of uncertainty. But we can and must, continually, explore the sensitivity of our estimates and our decisions to the gaps in our knowledge. We have been doing that and we will keep at it.

In summary, present reactors pose no undue risk to public health and safety. This policy statement acknowledges that and indicates a willingness to permit continued operation of existing reactors as well as to license new reactors. This policy statement has been studied intensively for over three years. It has been reviewed carefully and endorsed by the Advisory Committee on Reactor Safeguards. It has not been lightly considered nor lightly decided. I am confident that the Commission has enunciated a sound regulatory policy.

Dissenting Views of Commissioner Asselstine

Summary

The foremost risk to the public from the operation of nuclear reactors derives from core meltdown accidents which can, through the release of substantial quantities of radioactive materials, result in the injury and death of a catastrophic number of people. This policy statement, which establishes Commission policies on these severe accident risks, represents one of the most fundamental regulatory decisions ever made by this agency. This statement, together with three other related regulatory decisions, will chart the future course of this agency and the nuclear industry on nuclear safety issues for many years to come. The three other decisions are the Commission's decision on the acceptability of the severe accident risk at the two operating Indian Point plants, the development of a backfitting rule incorporating a substantial safety threshold for the imposition of new requirements together with heavy reliance on quantitative cost/benefit analyses, and the development of a provisional, and ultimately a final, safety goal with numerical standards for evaluating the acceptability of nuclear accident risk. Taken together, these four Commission actions will set the framework for deciding whether the NRC and the industry will pursue existing and future significant safety issues, whether further improvements in safety will be pursued for both existing and future plants, and how such decisions will be made.

Unfortunately, the first two of these decisions by the Commission lead me to conclude that we are on the wrong course. My views opposing the Commission's Indian Point decision were set forth in considerable detail in the Commission's written decision (see CLI–85–06), and I will not rehearse those views here. Suffice it to say that the Commission's unsubstantiated and overly optimistic assumptions on the long-term acceptability of the severe accident risk posed to the public by those plants have now been extended by this policy statement to cover all existing and future nuclear powerplants in this country. In my judgment, the Commission's action today fails to provide even the most rudimentary explanation of, or justification for, these sweeping conclusions. As a basis for rational decisionmaking, the Commission's severe accident policy statement is a complete failure.

Existing Plants

I see at least four fundamental flaws in the Commission's policy statement as it applies to existing plants. First, while the policy statement reaches a positive conclusion on the acceptability of the severe accident risk posed by existing plants, it fails to articulate what that risk is; it fails to identify the relevant technical issues evaluated in assessing the acceptability of that risk; it fails to explain how those technical issues were considered and resolved by the Commission in reaching its positive conclusion; and it fails to demonstrate

[4] See 10 CFR 2.764(f) and "Statement of Policy: Further Commission Guidance for Power Reactor Operating Licenses," 45 FR 85236, December 24, 1980.

the technical support for that conclusion based on scientifically accepted principles and methodology.

Absent a detailed discussion of the severe accident risk posed by existing plants and of the reasoning and scientific basis supporting the Commission's conclusion on the acceptability of that risk, that conclusion must be viewed as nothing more than an unsubstantiated assertion deserving of little weight.

Second, the Commission's policy statement fails to provide any explanation of the Commission's treatment of uncertainties in evaluating the risk of severe accidents. The absence of virtually any explanation of how uncertainties have been treated in this policy statement further undermines the validity of the Commission's broad conclusions on the acceptability of the risk posed by severe accidents.

Third, the Commission fails to address in a clear and consistent manner the need to prevent further severe reactor accidents. Although the Commission's policy statement pays lip service to this goal, it fails to include the means to fulfill that objective.

Fourth, the Commission's policy statement places undue reliance on probabilistic risk assessments (PRA's) as a means for resolving severe accident questions for existing plants. This reliance fails to recognize present weaknesses in these assessments due to the limited number of PRA's available thus far, the variations among the existing PRA's, the absence of accepted guidelines on how to conduct PRA's and to evaluate them in making severe accident risk judgments, and the uncertainties inherent in attempting to extrapolate plant-specific PRA results to other plants.

Future Plants

The Commission's policy statement is equally flawed in its treatment of severe accident risk for future plants. First, the policy statement promises that the Commission will make final decisions in the near term on the acceptability of new plant designs for severe accident purposes. At the same time, the policy statement acknowledges that key elements in evaluating the acceptability of severe accident risk—criteria for the preparation and evaluation of PRA's, containment performance criteria, and criteria for evaluating the risk contributions due to sabotage and human performance—will not be available for some time. Thus, the Commission's approach is to agree to make final decisions on severe accident risk for future plants before the technical basis for evaluating the nature and acceptability of that risk is available.

Second, the policy statement does not go far enough in insisting upon reductions in the severe accident risk of future plant designs. Such reductions are much more readily achievable in new designs for as-yet unbuilt plants than for existing plants. While the Commission's policy statement urges reactor designers to make safety improvements in the designs of future plants, it does nothing to require that improvements be made.

Third, the Commission's policy statement retains the option of authorizing the start of construction of future plants based upon only limited plant design information, including the limited design information which would be needed to support issuance of a preliminary design approval (PDA). Past experience with nuclear powerplant design, construction and regulation has taught us the many pitfalls of the old design-as-you-build approach. By continuing to allow the start of plant construction with only limited design work complete, the Commission seems committed to repeating the mistakes of the past—mistakes which have led to the deferral of significant design issues until the construction and pre-operation stages and the need to modify work already in progress or completed.

Taken together, these flaws in the Commission's severe accident policy statement cast doubt upon the adequacy of the Commission's overall approach to dealing with severe accident risk and undermine the validity of the Commission's sweeping judgments of the acceptability of that risk for existing and future plants.

Discussion

Before elaborating on the major infirmities of this policy statement, it is useful to explain what we know about the severe accident risks to the public.

Risks

Risks are commonly defined as the product of the probability that an event will occur and the consequences of the event happening. In regulating the nuclear industry, the Commission makes extensive use of a methodology called probabilistic risk assessment (PRA). In conducting a PRA the analyst calculates the core meltdown probability and, given a particular core meltdown scenario, the analyst then estimates the consequences to the public. The Commission uses the bottom line of these PRA's in deciding whether to improve reactor safety or to relax the safety standards even though such PRA's do not consider all contributors to core meltdown risks or quantify all of the uncertainties.

A typical result of a PRA which is used by NRC in reaching safety decisions is the estimated core meltdown probability of about one in ten thousand (or 10^{-4}) per reactor year. However this probability estimate is often based on what is called the "median" value. It is important to understand just what the meaning of this bottom line number really is. Because of major inadequacies in the data base, because of the vast complexity of nuclear plants, because a tremendous number of assumptions must be made in calculating core meltdown probabilities, and because large scale core meltdown phenomena are poorly understood, no one calculation will yield a remotely meaningful probability of catastrophic consequences. Therefore, the PRA analyst must perform thousands of individual estimates of the core meltdown probability while randomly varying within chosen distribution patterns which themselves are not precisely known individual component failure probabilities, human error rates, and theoretical models that are thought to describe most of the important physical processes or engineering behavior. Any one of these individual estimates is as likely to be valid as the estimate resulting from any one of the other thousands of calculations. There is a crucial, but untenable, underlying assumption that all core meltdown sequences have been accounted for in the estimates. The analyst then scans all of the estimates and picks the probability value at which half the estimates are above the half are below. This number is called the median. It is, according to the Commission, the "best estimate". When calculated in this way, however, one cannot say with any confidence that this median value is the true core meltdown probability. Nonetheless, the Commission airbitrarily chooses this median number to use in making its regulatory decisions.[1]

[1] The practice of using median estimates was strongly criticized by our Advisory Committee on Reactor Safeguards during its July 11, 1983 meeting with the Commission. The ACRS recommended that mean rather than median estimates be used, and noted that use of median rather than mean estimates can result in a substantial underestimate of the effects of uncertainties in making reactor accident risk estimates. As indicated above the median is that point on a spectrum at which half of the values fall above and half fall below. The mean is the average value of the spectrum of risks and is also called the "expected value."

Continued

The spread in the estimated core meltdown probabilities for a typical plant range from approximately one chance in one thousand (10^{-3}) per year to one chance in one hundred thousand (10^{-5}) per year, with a median value of one chance in ten thousand (10^{-4}) per year, give or take a few. However, there is no proof that the median of the calculated values reflects the actual risk any more than do the estimates of 10^{-3} per year or 10^{-5} per year.

Another typical result of PRA's is the prediction that about 1 out of 10 core meltdowns likely will result in lethal radiation doses to about 1,000 people. Such consequences of core meltdown accidents are attributable to degraded performance of the containment, which can come about in a variety of ways that are not precisely quantifiable. Because of these uncertainties in quantification, the fraction of core meltdown accidents which would lead to catastrophic consequences is actually a range of values. The range could be two or three times greater than the above estimate; or it could be two or three times less. Picking the minimum factor of 2 and assuming there are 100 operating reactors, the approximate range of chances of a catastrphic accident between now and the year 2000 would be anywhere between 0.2 (2 chances in ten) and 0.001 (one chance in a thousand).

Therefore, the information before the Commission indicates that there could be anywhere between a 20 percent chance and a 0.1 percent chance of an accident at a nuclear reactor in the next 15 years that would result in lethal doses to about 1,000 people. The range of chances could be larger than this if one considers all contributors to the core meltdown probability and all uncertainties. Likewise, the number of deaths could be larger or smaller. Admittedly, there are many ways of going about estimating the range of risks. However, if there is validated quantitative information on core meltdown risks that is better, it has not yet been demonstrated. Thus, because of the many uncertainties involved in calculating both the probabilities and the consequences of core meltdowns, one number does not give a true picture of the actual risk. A range of possibilities is a more accurate representation of our understanding of the issue.

A serious consideration of the core meltdown risks would consider this full range of calculated risks and would address forthrightly the question of whether this risk is acceptable or unacceptable, both for the immediate future and over the long term. The Commission's consideration of severe accident risks instead focuses on a median number, ignoring the actual range of of values and the uncertainties inherent in using a median number for decisionmaking.

Since the foremost risk to the public from the commercial nuclear industry derives from severe accidents, adopting a policy that seeks to resolve severe accident issues in a definitive manner is the most basic duty which can be undertaken by the Commission in meeting its responsibility to decide what constitutes acceptable risk to the public. The Commission claims in this policy statement to have examined an extensive range of technical issues relating to severe accideht risks in reaching its judgment "that existing plants do not pose an undue level of risk to the public." The Commission's policy statement does not, however, incorporate an explanation, or for that matter even a description, of the most significant issues that have been resolved and the manner in which they were resolved. *Nor* does it include a description of the methods of analyses used in resolving the issues or decision criteria that were used for reaching the ultimate judgment. It is, therefore, impossible to discern the bases for the Commission's decision.

Uncertainties

A paramount concern regarding the acceptability of the risks to the public that must be resolved is how to reach a judgment on this issue in the face of enormous uncertainties which are up to 100 times the median value used by the Commission. Depending on how such uncertainties are factored into the decision, judgments could range from requiring substantial efforts to reduce core meltdown risks to doing nothing about them. Scientifically accepted data and methodology are not available at this time to reduce substantially those uncertainties so that, as the technical staff of the NRC has repeatedly told the Commission, it is "mandatory" to consider them in any application of risk assessments.

After being informed of the uncertainties in the risk estimates, the Commission simply ignores them. The Commission fails to provide any basis for its decision to ignore these uncertainties. Absent some rational treatment of these uncertainties or a convincing justification for why they can be ignored, the public can have little confidence in the Commission's conclusion that the risks to the public from a severe accident at a nuclear powerplant are acceptable. The only available explanation of the NRC's approach to making decisions in the face of these significant uncertainties is given on pages 133 through 140 of NUREG–1070, "NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulation", October 1984. About half of the pages are blank and the remainder are not much better. This discussion of uncertainties is inadequate and fails to provide a sufficient basis to justify the Commission's sweeping conclusions on the acceptability of the severe accident risk.

Another fundamental issue requiring resolution is the level of risk to the public that reasonably should be found acceptable. Beyond making a sweeping conclusion that the severe accident risk at the existing plants does not pose an undue risk to the public, the Commission fails to address this fundamental question. In fact, the Commission's technical staff is just now embarking on a program of analysis that "will form part of the basis for a Commission judgment on the level of safety presently achieved by existing plants for severe accidents."[8] Since the Commission is just beginning this program, it cannot serve to justify the Commission's judgment on the acceptability of the severe accident risk.

In its Indian Point decision, the Commission adopted specific point estimates of core meltdown risks for the Indian Point reactors and found them to represent an acceptable level of risk. In the course of developing this policy statement the Commission expressed much interest in the bottom line results of *all* completed PRA's, whether the reported point estimates were the mean or median. The technical staff has repeatedly cautioned the Commission that such bottom line numbers are not credible. What then is the basis for the Commission's position that the level of severe accident risk posed by the existing plants is acceptable?

The Commission's decision-making process in developing this policy statement is simply to rely upon "point

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Some PRA analysts base their estimates on the mean. However, the Commission has twice endorsed use of the median value. The first time was when the Commission endorsed WASH-1400 (Reactor Safety Study) in 1975 and the second time was when the Commission approved the provisional Safety Goal Policy Statement (NUREG–0880, Revision 1) in 1983.

[8] See, NUREG–1070, "NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulation. October 1984, p. 27.

estimates" of the core meltdown risks without any consideration of the effects of the uncertainties. This approach can lead to a decision to doing nothing to reduce core meltdown risks. Factoring into the decision the uncertainties in estimating the level of core meltdown risks would lead to a decision to search for ways to reduce the risks. However, given the current political climate, there is little sympathy for backfitting existing plants. Thus, the Commission chooses to rely on a faulty number which supports the outcome they prefer and to ignore the uncertainties, those that are known and quantified and those that are not quantifiable.

What level of confidence does the Commission have in its judgment that core meltdown accidents present no undue risks to the public? The Commission nowhere expresses the degree of confidence it seeks to ensure that catastrophic accidents do not happen. Yet, the Commission's chief safety officer recently wrote: "In view of the large uncertainties surrounding methods of assessing severe accident risk, the *level of assurance* (or confidence) of no undue risk to the public is regarded as no less important than the estimated *level of risk* itself (emphasis in the original)." Letter from H.R. Denton, NRR, to A.E. Scherer, Combustion Engineering, Inc., dated December 28, 1984, subject "SECY-84-370, Severe Accident Policy".

Another problem with the Commission's policy statement is that it clearly contradicts what the Commission is doing in other areas. For example, in this policy statement the Commission states: "A fundamental objective of the Commission's severe accident policy is that the Commission intends to take all reasonable steps to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur." However, compare this statement with the Commission's proposed backfitting standard: "The Commission shall require the backfitting of a facility *only* when it determines, based on a systematic and documented analysis * * * that there is a substantial increase in the overall protection of the public health and safety * * * to be derived from the backfit *and* that the direct and indirect cost of implementation for that facility are justified in view of this increased protection." (emphasis added) The Commission has already defined a substantial increase in protection as meaning a backfit that would at least reduce the "point estimate" of the calculated core meltdown risks by half. Unless such a reduction can be "demonstrated", the Commission will not consider requiring the change. This is a much higher barrier to requiring improvement in reactor safety than the policy statement would have us believe is the Commission's policy.

Further, the Commission's provisional safety goal is not intended to regulate on the basis of preventing core damage accidents, as implied in the above purported fundamental objective. Rather, the safety goal assumes that the containment is an independent bulwark capable of limiting the external release of radioactivity to modest amounts for most core meltdown accidents. Thus, according to the Commission, there is no need to regulate on the basis of preventing core meltdowns. I am not as sanguine as the Commission on the acceptability of core meltdown accidents. Even if the containment happens to retain most of the radioactive fission products in the next severe accident, another accident equal to or more severe than that which occurred at Three Mile Island would be unacceptable to the public and the Congress and would be disastrous for the nuclear industry and the NRC.

But more importantly, the Commission's belief that the containment will retain all but modest amounts of radioactivity during most core meltdowns is not yet supportable based on scientifically accepted principles and methodology. There simply is no actuarial experience or direct experimental data on large scale core meltdown phenomena or containment performance characteristics given a core meltdown. In the past, estimates of the quantities of radioactive releases to the environment have been based on not much more than interpolations of extrapolations of approximations. It is for this reason the Commission has an ongoing program, which has cost a quarter of a billion dollars in the last few years, in an attempt to bring some science to estimating the core meltdown risks. However, even in this program the data being generated are from limited small scale tests.

Thus, a reading of this policy statement indicates that the Commission's claim that in developing this policy statement it has examined an extensive range of issues is incorrect. It shows rather that the Commission either examined the wrong issues or gave short shrift to the fundamental issues.

In failing to define accurately the level of severe accident risk at the existing plants and to address the need for additional changes to the plants to make this risk acceptable for the long term, the Commission is repeating past failures to deal effectively with the severe accident question. The concept of the reactor containment originally evolved as a vessel to contain a full core meltdown. But in the mid-1960's, the reactor designers began placing high powered cores into roughly the same kind of containment. The decay heat of those higher powered cores was so high that the containment vessel could no longer be considered as an effective independent barrier to the release of the fission products evolved during a core meltdown. At that time, the Atomic Energy Commission's Advisory Committee on Reactor Safeguards (ACRS) began urging the development and implementation, in about two years, of safety features to protect against a loss of coolant accident in which the emergency core cooling system did not work. The AEC and the industry believed that sufficient data were available to justify with a high degree of confidence the adequacy of the then-existing safety standards. Therefore, the AEC ignored the advice of the ACRS.

Over the years, the AEC and the NRC after it have reiterated these sweeping and optimistic statements on severe accident risk. At the same time, the numerous technical flaws in the Commission's judgments have become readily apparent as more information and data regarding the level of safety of the reactors has become available.[9]

When all of the available data are considered, I believe it fair to say that the estimated uncertainties in the risk calculations today are as large as they were at least ten years ago. Yet, the Commission is once again sweeping aside these uncertainties in order to make the same unsubstantiated and overly optimistic generalizations about the acceptability of the current level of severe accident risk which have been proven wrong in the past.

Needed Improvements

A disciplined approach to deciding whether to require core meltdown risk reduction measures should not only specify the Commission's expectations on addressing uncertainties but it should also describe the Commission's policy

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[9] Dr. David Okrent (who has been a member of the ACRS since 1963) has compiled a detailed account of the judgments made by the AEC and the NRC on severe accident risk and the technical flaws in those judgments. See David Okrent, *Nuclear Reactor Safety-On The History of the Regulatory Process*, University of Wisconsin Press, 1981 pp 163-178.

on acceptable ways to perform cost-benefit analyses.

Further, guidance from the Commission is needed on whether to emphasize core meltdown prevention measures or core meltdown mitigation measures. Of course, in order to develop a policy on the latter (whether for existing plants or future plants), one must first identify the root causes of core meltdown risks. One must also develop a policy on containment performance expectations.

Unfortunately, the Commission refuses forthrightly to address these issues. An effective guide to regulatory decision-making on the treatment of severe accident issues requires an understanding of what is expected by way of containment performance, of the root causes of core meltdown risks, and of the methods for performing sound cost-benefit analyses. Yet all of these elements are missing from the Commission's policy statement. The Commission's actual decision-making guidance in this policy statement is limited to the statement that a new requirement might be imposed if it involves "low-cost changes in procedures or minor design modifications."

The Commission claims that PRA's identify the plant specific vulnerabilities that dominate the core meltdown risks. It is true that PRA's can identify *some* of the vulnerabilities to catastrophic accidents. But the Commission's rationale for relying upon PRA's in assessing core meltdown risks begs the questions: what of the uncertainties in PRA's? What of oversights in the analyses? What of the multitude of assumptions and approximations in the PRA's? What of the residual risks once the specific vulnerability has been fixed? These questions are germane to resolving severe accident issues. Yet they are not addressed in the Commission's policy statement.

Operational experience gives additional insight into the level of safety. Actuarial experience with reactor accidents indicates that the average core meltdown frequency is not above the upper limit of the PRA results. Core meltdown accidents involve multiple failures and a progression of events that make close calls somewhat identifiable. If the industry average of the core meltdown frequency were as high as 10^{-3} per reactor year, one would expect more close calls on core meltdowns than appear to have occurred within the more than 800 reactor years of U.S. nuclear power experience. But such actuarial inferences must be made cautiously in part because the operating reactors continue to surprise us. What actuarial experience we have is severely limited by our lack of detailed understanding of the performance of the plants, their designs, their weak spots, and because of the wide variations in the designs and in utility capabilities. Further, the usefulness of actuarial experience in drawing broad conclusions about commercial nuclear reactors is highly controversial and fraught with uncertainties.

The Commission argues that credit can be taken for the improvements implemented to address specific close calls such as the TMI accident, the Browns Ferry fire and the Rancho Seco transient. Each of these were previously unrecognized (or at best inadequately appreciated) accident sequences. This is also true of, for example, the Susquehanna station blackout event from a single failure, the Indian Point vulnerability to a single failure of a battery, and the so-called interfacing system LOCA's for boiling water reactors. None of these latter events were identified or highlighted through PRA's nor were they expected to be, given the level of detail that typically goes into a PRA and given the subjective nature of PRA's. Whether these latter events should be called close calls is arguable but their occurrences certainly suggest a need to consider the root causes of significant operating events and the collective meaning of those events before passing judgment on the acceptability of the level of safety achieved at existing power reactors. Common sense also suggests completing such an analysis before developing guidelines for the design of future reactors. Yet all of these concerns are swept aside in the Commission's policy statement.

The TMI Action Plan called for a large number of modifications to the operating plants. In addition to those modifications, the Action Plan committed to a rulemaking to consider to what extent, if at all, existing nuclear power plants should be required to deal effectively with damaged core and core meltdown accidents. There was to be a demarcation between those plants already operating or under construction and the next generation of future plants. Because the Commission perceived in 1980 that there would be a long hiatus in new plant orders, ample time existed to reconsider the General Design Criteria, the design bases, and the other regulations in light of all that had been learned through the years of experience with large power reactors, including the TMI accident. From this in-depth assessment of the strengths and weaknesses of the large power reactor designs and the approach taken by utilities toward constructing the plants, NRC would then be in a position to articulate safety principles that it expected to be incorporated into designs for future applications. Thus, the Commission in 1980 signaled there would be a significant step forward in advancing the protection of the public. The Commission in this policy statement takes several steps backwards.

One backward step discussed above is the Commission's decision to accept the core meltdown risks as they exist in the current generation of plants without even addressing some of the most fundamental issues. Another backward step is abandonment of the expressed desire for a fresh look at light water reactor safety for future designs and the insistence on improvements in the level of severe accident risks for any future plants. A third backward step in this policy statement is the return to the philosophy of the 1960's and 1970's that construction permits can be issued based on only partial design information.

For any future reactor orders, nuclear utilities themselves have expressed a desire for plant designs that are simpler, safer, and more forgiving. Both the Electric Power Research Institute (EPRI) and Edison Electric Institute (EEI) have impressed on the Commission the need for a fresh look at light water reactor technology. These utility sponsored organizations have also indicated that plant construction for new plants should not begin until there exists an essentially complete design for the plant. Yet none of these forward thinking requirements are to be found in the Commission's policy statement. Instead, the Commission states that it will be satisfied with mere refinements in the old designs and that it is willing to continue to approve partial designs for issuance of Construction Permits.

I cannot leave this latter point without a sad commentary on the Commission's priorities. One issue in this policy that commanded great interest within the Commission was how to circumvent its regulation that requires a comparison of a design to the staff's Standard Review Plan. This effort was motivated by the objections of one reactor vendor. Indeed, the Commission's efforts to use this policy statement as a vehicle to permit the reactor vendor to circumvent the Commission's regulations took precedence over any Commission consideration of such fundamental issues as the actual level of severe accident risk to the public, the acceptability of that risk and potential measures to reduce that risk

A Rational Approach to Severe Accident Decisionmaking

What the Commission should have done in its policy statement is to set forth precisely and in understandable terms what our present estimation of the risk of severe accidents is, whether the Commission believes that risk to be acceptable or not, what specific technical support can be offered in support of that judgment, and how the relevant uncertainties have been treated. The Commission should also have come to grips with a central question in our regulatory program: that is, given our present state of knowledge concerning severe accident risks, should we continue to pursue possible improvements in severe accident prevention and mitigation? If the Commission does not believe that the present level of severe accident risk is acceptable for the remaining 40-year life of some existing plants, then the Commission should outline its program for bringing this long-term risk within acceptable bounds. Only through such a process can the technical community, other public policy makers and the public understand and accept the Commission's judgment on the severe accident risk question. Unfortunately, such an analysis is nowhere to be found in the Commission's policy statement.

Based upon the preceding discussion, I would have reached the following conclusion: First, the risk to the public posed by severe accidents at the existing plants is not acceptable for the full remaining operating lives of those plants. Therefore, the Commission should continue to pursue cost-effective risk reduction measures for these plants. I would apply the as-low-as-reasonably-achievable (ALARA) principle to reducing severe accident risk, subject only to the qualification that changes which would only result in trivial safety improvements need not be pursued. I would have simply acknowledged the obvious: that the public and the Congress will not tolerate, and the industry and the NRC cannot allow, another severe accident as serious as the Three Mile Island accident or worse. My views in this regard are identical to those expressed by the Kemeny Commission nearly six years ago:

> Whether in this particular case we came close to a catastrophic accident or not, this accident was too serious. Accidents as serious as TMI should not be allowed to occur in the future.
>
> The accident got sufficiently out of hand so that those attempting to control it were operating somewhat in the dark. While today the causes are well understood, 6 months after the accident it is still difficult to know the precise state of the core and what the conditions are inside the reactor building. Once an accident reaches this stage, one that goes beyond well-understood principles, and puts those controlling the accident into an experimental mode (this happened during the first day), the uncertainty of whether an accident could result in major releases of radioactivity is too high. Adding to this the enormous damage to the plant, the expensive and potentially dangerous cleanup process that remains, and the great cost of the accident, we must conclude that—whatever worse could have happened—the accident had already gone too far to make it tolerable.
>
> While throughout this entire document we emphasize that fundamental changes are necessary to prevent accidents as serious as TMI, we must not assume that an accident of this or greater seriousness cannot happen again, even if the changes we recommend are made. Therefore, in addition to doing everything to prevent such accidents, we must be fully prepared to minimize the potential impact of such an accident on public health and safety, should one occur in the future.

Report of the President's Commission on The Accident at Three Mile Island, p. 15.

In order to reduce the severe accident risk over time to acceptable levels, I would have undertaken four specific initiatives. First, I would have required a detailed search for plant-specific equipment and design vulnerabilities at each existing plant to identify and correct those weaknesses which constitutes significant contributors to the risk of a severe accident.

Second, I would have initiated a concerted effort to improve operational performance at the existing plants, with special emphasis on areas of weakness throughout the industry (maintenance and surveillance testing stand out as good examples) and on specific utilities with a history of marginal performance. The June 9, 1985 operating event at the Davis Besse nuclear powerplant once again demonstrated the dangers inherent in the combination of a marginal plant design and a utility with marginal operating performance.

Third, I would have initiated a comprehensive assessment of the level of safety and the existing plants have achieved. The object of this effort would be to identify the root causes of severe accident risks. This effort would also identify possible measures which offer the promise of significantly reducing severe accident risk by overcoming the adverse effects of equipment breakdowns, human error, design deficiencies and areas of present uncertainty which are likely to persist despite our best efforts to address my first two initiatives. Indeed, as the Commission's chief safety officer noted in a June 27, 1985 memorandum to the Executive Director for Operations:

> I believe that the recent Davis-Besse event illustrates that, in the real world, system and component reliabilities can degrade below those we and the industry routinely assume in estimating core melt frequencies. Our regulatory process should require margins against such degradation and also to reflect the uncertainties in our PRA estimates.

Finally, for future plants, I would have explicitly required measures to improve the margin of safety against severe accidents in future plants and to address the mistakes of the past. Such measures could include requirements for greater simplicity in plant design, improved maintainability, and a requirement for essentially complete plant designs prior to the issuance of NRC approval for the start of plant construction.

I believe that these measures would be sufficient to bring the risk of severe accidents within acceptable bounds for the remaining operating lives of the existing plants and for the operating lives of any future plants. Moreover, such an approach would do much to restore public confidence in nuclear power and in the effectiveness of the NRC's regulatory process. It is unfortunate that the Commission has chosen another path. However, key decisions remain to be made by the Commission in adopting a final backfitting rule and a final safety goal. Those decisions represent a final opportunity to come to grips with many of the pivotal issues avoided in this policy statement. In that regard, it is encouraging that there appears to be an emerging consensus within the NRC senior technical staff and within the ACRS in favor of safety improvements to reduce severe accident risk both for existing and for future plants.

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NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Severe Accident Design Criteria; Withdrawal of Advance Notice of Proposed Rulemaking

AGENCY: Nuclear Regulatory Commission.

ACTION: Withdrawal of advance notice of proposed rulemaking.

SUMMARY: The Nuclear Regulatory Commission (NRC) is withdrawing an advance notice of proposed rulemaking (ANPRM) entitled "Severe Accident Design Criteria," because the issues addressed in this ANPRM are being handled in a Policy Statement entitled "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," published elsewhere in this issue.

DATE: This advance notice of proposed rulemaking is withdrawn effective August 8, 1985.

FOR FURTHER INFORMATION CONTACT: Miller B. Spangler, Special Assistant for Policy Development, Division of Systems Integration, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: 301-492-7305.

SUPPLEMENTARY INFORMATION: On October 2, 1980, the NRC published an ANPRM entitled "Severe Accident Design Criteria" (45 FR 65474). It was subsequently decided to handle this issue in a Policy Statement. The Policy Statement, entitled "Proposed Commission Policy Statement on Severe Accidents and Related Views of Nuclear Reactor Regulations," was published for comment on April 13, 1983 (48 FR 16014). After consideration of the comments, the NRC has issued a final Policy Statement entitled "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" which appears elsewhere in this issue. Consequently, this serves notice of the withdrawal of this ANPRM.

Dated at Washington, DC this 5th day of August 1985.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

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