BROOKHAVEN NATIONAL LABORATORY
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Upton, Long Island, New York 11973

Department of Nuclear Energy

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FTS 666

March 1, 1985

Mr. George Thomas
Reactor Systems Branch
Division of Systems Integration
U.S. Nuclear Regulatory Commission
Mail Stop P-1132
Washington, D. C. 20555

Dear George,

I have enclosed the revised list of information items for the containment response review which you requested (Attachment 1). I have included a schematic of the secondary containment (reactor building) indicating the expected failure location (see Appendix M of the Shoreham PRA) and the water relocation for gross overpressure failures (Attachment 2).

The previously transmitted BNL decontamination factors are included as Attachment 3. These are time-averaged pool scrubbing factors based on SPARC (NUREG/CR-3317) and do not take credit for in-vessel retention and primary containment hold-up. Note that for the Class-I sequences the suppression pool is subcooled and that the BNL decontamination factors are high and in substantial agreement with the Shoreham PRA results. However, for Class-IV sequences, the pool is heated to saturation before core melt and the BNL decontamination factors are much lower. For the Class-IV ATWS with failure at the basemat (γ''), it is assumed that the pool is relocated to the annular region of the reactor building which surrounds the primary containment (see Attachment 2). Thus, the in-vessel release through the SRV's see the same scrubbing as the ex-vessel release through the vent pipes. The slight difference in the decontamination factors (9 compared to 14) for the two releases depend on gas blowing rates at the time the scrubbing occurs.

If you have any questions, please call.

Very truly yours,

Ken

Kenneth R. Perkins

KRP:tr
Attachments

cc: W. Y. Kato (w/attachments)
R. A. Bari " "
W. T. Pratt " "
J. Rosenthal (w/attachments)
M. Wohl " "
F. Eltawila " "

D/35

ATTACHMENT 1

REQUEST FOR INFORMATION

1. Table II of Appendix M gives different pressure limits for the longitudinal reinforcement bars at the base of the containment and in the wetwell region. However, the longitudinal bars appear to be continuous and should therefore have the same stress. Please explain the basis for the different results.

2. Table II of Appendix M indicates that the shear bars at the base and drywell head have the lowest pressure holding capability (121 psi and 120 psi, respectfully) but the discussion indicates that the additional reinforcement will preclude this failure mode. Since the containment failure mode is a key ingredient of the release estimates, please provide a quantitative estimate of the additional shear strength provided by the non-shear reinforcement bars.

3. If shear failure is precluded as discussed in Section 3.2 of Appendix M, "it appears that the ultimate capacity is controlled by the yield of the longitudinal and the hoop bars at about 123 psi." These two failure modes appear to be very important to subsequent fission product release (particularly for Class IV ATWS) since they will both occur in the wetwell region. Please provide an estimate of the size, location and direction (vertical or horizontal) containment failures for each of the three possible failure modes.

4. Section 3.6 of the PRA takes credit for containment leakage which will prevent gross containment failure for all pressurization rates except the very rapid pressurization associated with large breaks. However, the structural analysis by Stone and Webster (Appendix M) did not identify any significant source of leakage. The basis for the expected leakage source and the leakage rate as a function of pressure should be provided.

5. The basis for the partitioning between release category 10 and 11 (no pool bypass vs. partial pool bypass) should be provided. The phenomenological basis for the estimate of only 10% bypass should be provided. Preliminary results from IDCOR indicate that for some BWR sequences the vessel will fail with only 20% of the core molten. Presumably 80% of the melt release would bypass the SRV's and be released into the drywell.

6. The basis for binning into release categories is poorly described and the transfer from Tables H.4-8 etc. into the 16 release categories is difficult to interpret. A table listing the specific sequences which are binned into each category should be provided.

7. The lack of R_5 sequences in the release categories makes it apparent that these releases have been binned "downward" into the lesser release category R_4. The basis for this "downward" binning and any other sequences that are moved to less severe categories should be provided.

8. Table H.4-25 appears to be incomplete in that it does not include sequences D6 and D8. The completed table should be provided.

9. The source escape fractions used for end state screening (Table 3.6-10) appears to be quite arbitrary yet it greatly influences the importance ranking. In particular: the use of I as the surrogate for melt release ignores the fact that there are noble gases in the melt release which will not be scrubbed at all; the use of a large scrubbing factor (500) for C_4 transients is inappropriate since most of the melt release will be released directly to a failed containment; the reduction factor of 0.01 for γ'' failures is indefensible since the event tree precludes everything but large ruptures where the pool will be blown out into the reactor building at high pressures.

   Table 3.6-10 should be replaced by a table with defensible reduction factors. As a minimum the table should include a separate category for C_4 transients, which recognizes the defined sequence of events (containment failure before core melt). In addition, a detailed justification for each reduction factor should be provided along with the numerical results of the ranking process. This revised table will provide the basis for our independent importance ranking based on revised estimates of accident frequency and reduction factors.

10. Sheet 1 of Figure H.4.2 has been reduced so that it is illegible. A full-size legible copy should be provided.

11. Appendix L provides a detailed discussion of the disposition of the corium (90% is expected to go down the vent pipes) based on the revised reactor pedestal geometry illustrated in Figure L.3-2. However, this figure is inconsistent with other descriptions of the geometry (e.g., Figure 2.3-2) and provides inadequate information for an independent assessment of the corium disposition. Please provide detailed (as built) drawings of the vent pipes and their covers within and external to the reactor pedestal region. Include a description of whether the air ducts and manways in the reactor support wall will be blocked during operation.

12. Provide the estimate of the fraction of the molten corium which is expected to spread out of the pedestal area through the open manways and air ducts in the reactor support wall.

ATTACHMENT 2

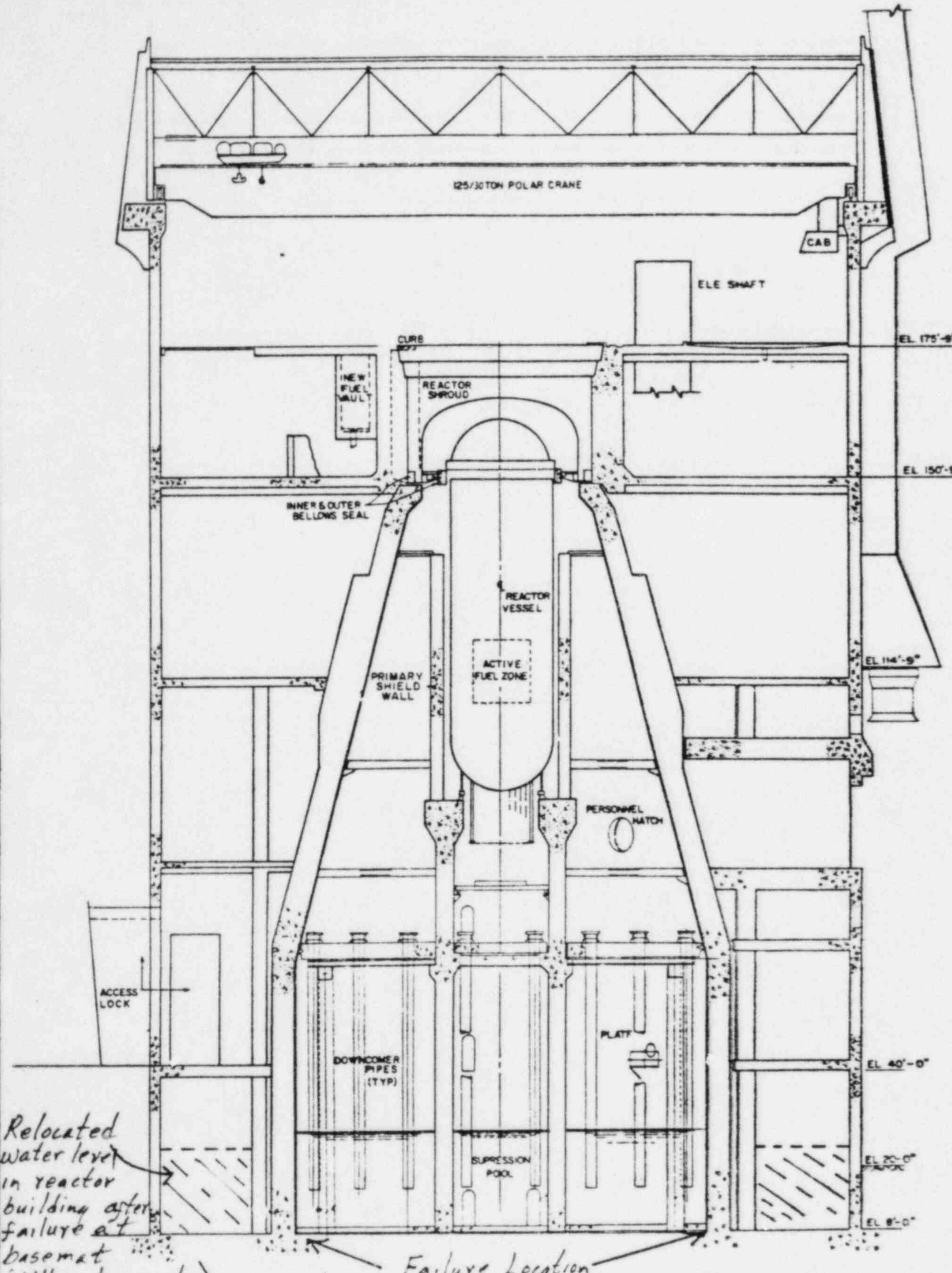

FIG. 6.2.1-4
PRIMARY AND SECONDARY CONTAINMENT
SHOREHAM NUCLEAR POWER STATION - UNIT I
FINAL SAFETY ANALYSIS REPORT

Note that 16% of pool is estimated to be vaporized during blowdown of primary containment after γ" failure

ATTACHMENT 3

Table 1 Comparison of Suppression Pool Decontamination Factors for Core Melt Accidents in Shoreham

Sequence	Decontamination Factors			
	Shoreham		BNL	
	SRV	Vents	SRV	Vents
TQUV (Class I-γ)	600	100	1000	75
ATWS (Class IV-γ')	600	100	50	22
ATWS (Class IV-γ'')	600	100	9	14

Felton

POLICY ISSUE
(Affirmation)

September 19, 1984

SECY-84-370

For: The Commissioners

From: William J. Dircks, Executive Director for Operations

Subject: NUREG-1070, "NRC POLICY ON FUTURE REACTOR DESIGNS: DECISIONS ON SEVERE ACCIDENT ISSUES IN NUCLEAR POWER PLANT REGULATIONS"

Purpose: To obtain Commission approval of a notice of policy.

Category: This paper covers a major policy on licensing and regulation of nuclear power reactors.

Issue: Coordination of final decisions on severe accident issues regarding future and existing plants with special focus on rulemaking for Commission certification of new standard plant designs.

Discussion: On April 13, 1983 the Commission approved and issued for public comment a "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation" (48 FR 16014). Twenty-six letters of comment were received. Since then, numerous meetings were held with the ACRS Class 9 Subcommittee and the full committee of the ACRS to exchange views on issues relating to the further development of severe accident policy for both future and existing plants. Working sessions were also held with representatives of the Industry Degraded Core Rulemaking program (IDCOR), which is sponsored by the nuclear utility industry to develop the technical basis for determining whether changes in regulatory requirements are needed to reflect severe accident considerations. The meetings between IDCOR and the staff focused on the definition of the most important technical issues of relevance to severe accidents and to compare IDCOR's independent models and assessments of severe accident behavior with work sponsored by the NRC.

Contact:
M. B. Spangler, NRR/DSI
492-7305

4pg

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D/2

A copy of a working draft of NUREG-1070 containing a revised severe accident policy statement, dated April 18, 1984, was sent to you for your information on May 7, 1984. Copies of this draft were placed in the PDR and sent to the ACRS for its review. The April draft departed substantially in form and content from the proposed Severe Accident Policy Statement of April 13, 1983 in the following key respects:

(1) It includes a sharpened focus of the Policy Statement on future reactor designs and sets aside severe accident rulemaking for existing plants unless new safety information should develop information exposing severe accident vulnerabilities that would warrant rulemaking (see Chapter III of Enclosure 2).

(2) It is structured for the concurrent publication of the NUREG report that provides an expanded discussion of numerous interrelated ongoing severe accident programs. These include: the Severe Accident Research Program; the Source Term Program; the development of Safety Goals and the PRA Reference Document; the resolution of USIs and Generic Safety Issues; and the integration of insights from IDCOR, foreign reactor and regulatory experience as well as the staff review of new reactor designs (see Chapter IV).

(3) It provides an overview of, and staff response to, public comments and the views and recommendations received from the ACRS (see Chapter V).

(4) It includes a short appendix on the treatment of uncertainty in the severe accident program and a more detailed appendix on current information bearing on the need for generic design changes or further regulatory changes affecting nuclear power plants. The latter provides a rationale for the differential policy treatment of existing and future plants and an up-to-date information base to support a number of critical premises or assumptions underlying the basic strategies of the Policy Statement.

The enclosed draft of NUREG-1070 differs only a little from the April draft, principally through the inclusion of staff responses to the ACRS letter of July 18, 1984 (see pp. 38-43) and the various sections of the Policy Statement and NUREG where changes were necessitated by these responses. Other lesser changes were made to revise the draft in accordance with additional information from current programs.

Recommendation: That the Commission:

1. Review and approve the Federal Register Notice on "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" (Enclosure 1).

2. Note:

   a. The *Federal Register* notice of the Policy Statement will be published concurrently with NUREG-1070, "NRC Policy On Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulations" (Enclosure 2).

   b. Copies of the *Federal Register* notice of the Policy Statement and NUREG-1070 will be distributed to all Commission licensees and public commenters on the proposed Policy Statement of April 13, 1983. Single copies of these documents will be sent to other interested parties upon request.

   c. Prior to publication of the Policy Statement in the *Federal Register*, a public announcement will be prepared and a letter will be drafted to notify appropriate Congressional Committees of these Commission and staff actions.

Scheduling: It is recommended that the Commission consider approval of the Policy Statement in an open meeting. A staff review of severe accidents for the GESSAR II design is nearly complete in preparation for a rulemaking to certify that design for future reference. Also, the staff has been involved with the pretendering review of an application for Westinghouse Electric Corporation's advanced pressurized water reactor design RESAR-SP/90. Accordingly, early Commission approval of the

Policy Statement is desired to facilitate staff review of these designs and preparation for implementing this policy.

William J. Dircks
Executive Director for Operations

Enclosures:
1. Federal Register Notice of Commission Policy Statement
2. NUREG-1070 (Commissioners, SECY, OGC & OPE only)

Commissioners' comments or consent should be provided directly to the Office of the Secretary by c.o.b. Friday, October 12, 1984.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Friday, September 28, 1984, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

This paper is tentatively scheduled for discussion at an Open Meeting during the Week of October 8, 1984. Please refer to the appropriate Weekly Commission Schedule, when published, for a specific date and time.

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Enclosure 1

FEDERAL REGISTER NOTICE OF PUBLICATION OF

COMMISSION POLICY STATEMENT

ON

SEVERE REACTOR ACCIDENTS REGARDING FUTURE

DESIGNS AND EXISTING PLANTS

August 23, 1984

FEDERAL REGISTER NOTICE FOR COMMISSION POLICY STATEMENT ON ISSUES FOR NEW STANDARD REACTOR DESIGNS AND SEVERE ACCIDENTS

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. It's main focus is on the criteria and procedures the Commission intends to use to certify new standard 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). It also serves as notice of withdrawal of the advanced notice of proposed rulemaking, "Severe Accident Design Criteria," published on October 2, 1980 (45 FR 65474).

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 analyses and 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 this action the Commission hereby serves notice of the withdrawal of that 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), 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 safety and property and sees no present basis for immediate action on generic rulemaking or other regulatory changes for these plants because of severe accident risk. However, 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 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, safety and property.

Moreover, the Commission has ongoing nuclear safety programs that include: the resolution of new and several other Unresolved Safety Issues and Generic Safety Issues; 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.

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)];

- 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;

- 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, safety, and property;* 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 in future construction permit applications will be reviewed under the guidelines identified for 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. Single copies of NUREG-1070 also will be available upon written request and at no cost. Requests should be made to the NRC-GPO Sales Program, Attention: Sales Manager, Division of Technical Information and Document Control, U.S Nuclear Regulatory Commission, Washington, D.C. 20555 (Phone (301) 492-9530). Copies also may be purchased from the NRC-GPO Sales Program and the National Technical Information Service, Springfield, Virginia 22161.

The authority for this document is (Sec. 161, Pub. L. 83-703, 68 Stat. 948, as amended (42 U.S.C. 2201).

Dated at Washington, D.C. this _________ day of _________ 1984.

For the Nuclear Regulatory Commission.

____
Samuel J. Chilk,
Secretary of the Commission.

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*This criterion has an antecedent in Chapter 10, Sec. 103b of the Atomic Energy Act of 1954, which states, as a regulatory objective, "to protect health and to minimize danger to life or property."

POLICY STATEMENT ON SEVERE REACTOR ACCIDENTS REGARDING FUTURE DESIGNS AND EXISTING PLANTS

A. Introduction

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 focus on severe accident issues is prompted by the knowledge that accidents of this class, which are beyond the substantial coverage of design basis events, could constitute the major risk to the public associated with nuclear power plants. 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* with special focus on procedures for the certification of new standard plant designs. In line with its legislative mandate to ensure that nuclear power plants should pose no undue risk to public health, safety, and property, 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 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, safety, and property.

The main purposes of this Policy Statement follow:

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

*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.

- 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.

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 of the GESSAR II design is nearly complete in preparation for a rulemaking to certify that design for future reference. 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 realize safety and economic benefits, such as those achieved through greater simplicity; reduced size and generating capacity; 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, safety, and property; 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 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 staff and Commission approval. It is recognized, however, that a new design can go through different stages or levels of approval before receiving this final staff and Commission approval. 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 encouraged to ensure that the intent of the safety requirements are 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 of aerosols on engineered safety features. Following the outcome of severe accident research, 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 offsite property. However, the implementation of requirements to control occupational exposure should be considered along with the relatively small effects on public health and offsite

property 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. Further 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 with their broad ranges of uncertainty might 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, safety and property damage, 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. 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 a final design. Correspondingly, the staff will issue a Preliminary Design Approval and a Final Design Approval. A PDA is not, however, a prerequisite for an FDA. An applicant has the option to 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 final 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 questions. 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 a 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. Once approved by the staff, each reference final design will be subject to Commission approval by rulemaking. 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.

b. Certification of Reference Designs Previously Granted an FDA

In 1983, the NRC staff issued two Final Design Approvals for reference designs. These design approvals 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.

(1) Each reference design applicant with an existing FDA must perform an evaluation of its design against the current revision of the Standard Review Plan in accordance with 10 CFR 50.34(g). Upon submittal of this evaluation, the staff will amend the existing FDA to permit the design to be referenced in new CP and OL applications.

(2) The reference design must satisfy each of the criteria stated in III.B.2 before certification of the design. Each design approved by the staff for incorporation by reference in new CP and OL applications shall be subject to Commission approval by rulemaking. If a comprehensive PRA cannot be completed owing to the limited scope of the design, the applicant shall perform supplementary risk analyses, to the extent practical, in support of the rulemaking process. Failure to support the rulemaking in a timely manner can be cause for the staff to revoke the applicant's FDA. Also, as noted above, the limited scope of plant design and PRA analysis would lead to a partial loss of benefits relative to a certified design 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.

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 III.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 a 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 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 80 nuclear plants and expects to process applications to license another 40 or 50 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.

Since the accident at TMI, many changes have been implemented in existing plants resulting from recommendations of special inquiry groups, the TMI Action Plan (NUREG-0660 and NUREG-0737), and other information arising from NRC- and industry-sponsored research along with failure data from construction and operating experience. In addition, the NRC/AEC has sponsored eleven plant-specific PRAs and the industry has sponsored about as 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, safety, and property and dismisses therefore the need for prompt action on generic rulemaking or other regulatory changes for these plants. However, 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.

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.

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 <u>unique</u> characteristics of a specific plant design and, therefore, are dealt with through plant-specific modifications of relatively modest cost rather than major <u>generic</u> 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 design changes that are identified as necessary for public health and safety and for adequate protection of property 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 safety and property.

- 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.*

- Recognizing that plant-specific PRAs have yielded valuable insights 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.)

- The Commission intends to reserve for its own deliberations the resolution of severe accident issues affecting plants under construction. Therefore, individual licensing proceedings are not appropriate forums for a broad examination of the Commission's regulatory requirements relating to control and mitigation of accidents more severe than the design basis. Similarly, notwithstanding the Class 9 accidents review requirements for environmental hearings of the Commission's Statement of Interim Policy on "Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969" (45 FR 40101, June 13, 1980), the capability of current designs or procedures (or alternatives thereto) to control or mitigate severe accidents should not be addressed in case-related safety hearings.

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

INFORMATION REPORT

October 12, 1984 SECY-84-395

For: The Commissioners

From: William J. Dircks, Executive Director for Operations

Subject: UPDATE ON ACCIDENT SOURCE TERM REASSESSMENT

Purpose: To inform the Commissioners of the current status of the staff's efforts to reevaluate accident source terms.

Category: This is an information paper.

Background: In 1983 the Accident Source Term Program Office was formed in the Office of Nuclear Regulatory Research to focus and direct the staff's efforts to reassess the basic assumptions and the methodology for quantitative assessment of the releases of radionuclides resulting from core damage accidents (SECY-83-219). The clear need for thorough and extensive peer review resulted in a grant to the American Physical Society (APS) for a formal, broad-based review of the underlying scientific bases, in addition to the expert peer review of models and calculations, as described in SECY-83-219A. An informal status report was provided in my memorandum of June 19, 1984. This paper describes the current status as we approach the completion of this work and includes the schedule for completion and publication of the efforts of the Accident Source Term Program Office.

Discussion: Status of Contractor Assessments

The primary contract for improved source term estimates was placed with Battelle Columbus Laboratories (BCL). Several accident sequences for five reactors were selected for detailed analysis. The major selection criteria were the type of reactor, the type of containment, and a spectrum of accidents spanning the range of conditions considered important to source term estimation. The results of these calculations are described in a multiple-volume report known as Radionuclide Release Under Specific LWR Accident Conditions, BMI-2104. The series has seven volumes, with the following status:

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Volume	Typical Reactor	Type of Containment	Report Status
1	Surry	PWR-Large, Dry Containment	Draft Published
2	Peach Bottom	BWR-Mark I Containment	Draft Published
3	Grand Gulf	BWR-Mark III Containment	Draft Published
4	Sequoyah	PWR-Ice Condenser Containment	Draft Published
5	Surry (recalculated)	PWR-Large, Dry Containment	Draft Published
6	Zion	PWR-Large, Dry Containment	Draft Published
7	Reply to comments on technical issues		In Preparation

These formal draft reports are in bound version and have been widely distributed.

The computer codes developed to calculate the source terms reported in these volumes have been reviewed extensively, including an assessment of validation by the Oak Ridge National Laboratory (ORNL). The ORNL reviews are being published as a contractor report titled Review of the Status of Validation of the Computer Codes Used in the Severe Accident Source Term Reassessment Study (BMI-2104) and is available now in draft form.

An effort to quantify the uncertainty associated with source term estimates is underway at Sandia National Laboratory (SNL). The results will be published as a multiple-volume report with the title, Uncertainty in Radionuclide Release Under Specific LWR Accident Conditions. The status of this report is as follows:

Subject		Status
Volume 1	Executive Summary	In Preparation
Volume 2	PWR - Surry TMLB Sequence	Draft for comment
Volume 3	PWR - Surry S_2D Sequence	Draft for comment
Volume 4	BWR, Mark III - Grand Gulf TC Sequence	In Preparation

Status of the Peer Review

Upon completion, the NRC contractor source term reevaluation effort will have been reviewed at several levels by a number of organizations, including a panel of technical experts in an NRC-sponsored peer review group and an independent review by the American Physical Society Study Group (APS). The expert peer

review group consisted of about 15 technical experts from industry, universities, and technical organizations in the U.S. and abroad. The APS Study Group consists of 19 physicists and chemists from scientific organizations and academia. These scientists are conducting a broad-based review of the scientific bases underlying the source term re-analysis. The current status of these reviews is discussed below.

As the Battelle Columbus Laboratory and supporting laboratories developed the methodology for source term estimates, the peer review group assessed this work. Invited observers from academia, the industry, and the public were given time to comment on the analyses as well. Five meetings were held during January 1983, May 1983, July 1983, October 1983, and January 1984. Transcripts of the meetings were kept. The peer review group and the invited observers were asked to submit written comments concerning the technical basis, the methodology, and the specific analyses performed by Battelle.

The APS Study Group is currently conducting a broad scientific review of the source term analyses through an NRC grant issued in September 1983. The review covers not only the Battelle calculations but also includes the supporting source term research at Sandia National Laboratory, Oak Ridge National Laboratory, the Marviken Facility, the Power Burst Facility, EPRI-sponsored research and the industry-sponsored IDCOR calculations.

The Study Group's work includes: (1) critical review of experimental and theoretical work that has already been completed on this question, conducted with the support of the U.S. Nuclear Regulatory Commission, including analyses that have been performed for specific reactors; (2) evaluation of the extent to which more accurate estimates of radionuclide releases are now possible; (3) identification of areas in which the scientific basis for such estimation is now inadequate; and (4) indication of the degree to which presently planned or additional research efforts can be expected to rectify any inadequacies.

The findings of the Study Group will be discussed in a report to the American Physical Society and published in The Review of Modern Physics. The anticipated date for making the findings public is January 1985, at the APS meeting in Canada. However, the results may be released prior to then at a special press conference in New York City. This schedule is tentative; it depends on the progress that the Study Group makes towards finishing the report. ASTPO has and will continue to make every effort to assist the Study Group.

Status of Staff's Assessment

The staff's assessment and conclusions concerning the contractors' products will be contained in a NUREG report (NUREG-0956). This report will take into account not only the extensive analyses and evaluations of our contractors, but also the deliberations and written comments of the expert peer reviewers, information from similar or related efforts by other organizations (e.g., IDCOR, ANS) and from research conducted abroad. The report will also integrate the efforts of two separate Working Groups of experts inside and outside NRC, formed in response to early recognition that the behavior of the containment dominates accident source terms, i.e., the Containment Loads and Containment Behavior Working Groups.

The primary purpose of NUREG-0956 will be to assess the extensive body of new information with respect to its validity and potential applicability to the regulatory process, particularly in view of uncertainties still associated with the BMI-2104 methodology. Representative calculations for a number of accident sequences for different reactor types and containments will be summarized and reviewed with respect to the potential impact of the results on the estimates of the risks associated with severe reactor accidents.

This report (NUREG-0956) is presently in preparation. The major milestones for its completion and publication are shown in Table 1.

Please note that this proposed schedule includes a review by the ACRS, followed by a presentation to the Commission on or about March 27, 1985, prior to publication for public comment.

William J. Dircks
Executive Director for Operations

Table 1

Major Milestones for Completion and Publication of NUREG-0956

Report of APS Study	January *, 1985
Incorporation of Comments and Amendments as a result of APS report	February 18, 1985
Transmittal to ACRS for review	February 25, 1985
Commission Paper prepared	March 13, 1985
Commission Meeting, on or about	March 27, 1985
Publication of Draft NUREG-0956 for Public Comment	March 27, 1985
60-Day Comment Period Complete	May 28, 1985
Incorporation of Comments and Ready for Publication	June 1985

*APS estimate of completion. Dates which follow rely on receipt of APS Study in January.