February 28, 1986

POLICY ISSUE
(Information)

SECY-86-76

For: The Commissioners

From: Victor Stello, Jr., Acting
Executive Director of Operations

Subject: Implementation Plan for the Severe Accident Policy Statement and the Regulatory Use of New Source-Term Information

Purpose: To inform the Commissioners of the Staff's plans for implementing the Severe Accident Policy Statement and the regulatory use of new source-term information. The implementation plan provides for the resolution of severe accident issues through: (1) the systematic examination of existing plants for particular vulnerabilities to severe accidents, (2) the clarification of procedures and requirements for new-plant applications concerning severe accidents, and (3) the regulatory utilization of improved information on source terms.

Discussion: The implementation program incorporates 3 major elements. The first element is to formulate an integrated, systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant risk contributors that might be plant specific and might be missed absent a systematic search. The examination will pay specific attention to containment performance in striking a balance between accident prevention and consequence mitigation. The systematic approach will include 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. The licensee examinations may use a method developed by the IDCOR (Industry Degraded Core Rulemaking) program. As part of the NRC review of the licensee's examinations, any individual plant vulnerabilities will be identified together with the most cost-effective options for reducing vulnerabilities. The Commission's backfit policy will be used

Contact:
T. Speis, NRR
Z. Rosztoczy, NRR
49-28016

8603210217 XA

to decide which options need to be implemented. Any generic design changes that are identified as necessary for public health and safety will be required through rulemaking.

The tasks of this element are: (1) to evaluate the methodology developed by IDCOR and (2) to issue a generic letter to licensees containing guidelines and criteria for the systematic safety examination for individual plants. The program element is summarized in Figure 3.1 of the enclosure.

The second major element is to develop guidance on the role of PRAs in the approval of new applications. The NRC staff will use the safety insights gained from review of past PRAs to issue guidance on the form, purpose and role that PRAs are to play in severe accident analysis and decisionmaking for future plant designs and what minimum criteria of adequacy PRAs should meet. The elements in the guidance will include (1) the combinations of deterministic requirements and probabilistic considerations appropriate as bases for severe accident decisions, (2) the definition of the minimum content of the PRAs, and (3) the criteria for the regulatory interpretations of results from the PRAs. The program element is summarized in Figure 4.1 of the enclosure.

The third major element is the modification of our rules, guides and other regulatory practices not only to reflect those changes in our scientific understanding arising from our present and continuing research effort in severe accident releases ("source terms"), but also to reflect additional insights arising from severe accident research. While severe-accident phenomena research is still underway, the staff intends to initiate changes as soon as the available information warrants such changes, and is considering, herein, several such changes. The program element is summarized in Figure 5.1 of the enclosure.

Throughout the planned implementation of the Severe Accident Policy Statement, the resultant implementation will be influenced and constrained by reasonable treatments of (1) the generally large uncertainties associated with the PRAs' numerical assessments and (2) the continuing research on severe accident phenomenology.

The implementation program milestones rely upon the completed studies from past efforts and future collegial efforts by both the NRC staff and industry groups. The interfaces among these efforts are important for completing the program on schedule and are discussed specifically in Section 6 of the enclosure.

Victor Stello, Jr., Acting Executive
Director for Operations

Enclosure:
Implementation Plan for the
Severe Accident Policy Statement
and the Regulatory Use of New
Source-Term Information

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Enclosure

IMPLEMENTATION PLAN FOR THE SEVERE ACCIDENT POLICY STATEMENT AND THE REGULATORY USE OF NEW SOURCE-TERM INFORMATION

Office of Nuclear Reactor Regulation

U.S. Nuclear Regulatory Commission

February, 1986

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1. Background

On August 8, 1985, the U.S. Nuclear Regulatory Commission issued a policy statement on severe accidents[1]. The policy statement provides criteria and procedural requirements for the licensing of new plants, and sets goals and a schedule for the systematic examination of existing plants. On the basis of available information the Commission concluded that existing plants pose no undue risk to the public and the Commission sees no present basis for immediate action on generic rulemaking or other regulatory changes for these plants because of severe accident risk. Thus, the Commission withdrew the advanced notice of proposed rulemaking on Severe Accident Design Criteria published on October 2, 1980[2]. However, the Commission emphasized that systematic examinations of existing plants are needed, encouraged the development of new designs that might realize safety benefits, and stated that the Commission intends to take all reasonable steps to reduce the chances of occurrence of a severe accident and to mitigate the consequences of such an accident, should one occur.

With respect to new plant applications the Commission specified acceptance criteria and procedural requirements, which include completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes.

Within 18 months of the publication of the policy statement, the staff will issue guidance on the purpose, the form and the role that PRAs are to play in severe accident analysis and decision making for future plant designs and will issue criteria for the PRAs. The guidance will include (1) the combinations of deterministic requirements and probabilistic considerations appropriate as bases for severe accident decisions, (2) the definition of the minimum content of the PRAs, and (3) the criteria for the regulatory interpretations of results from PRAs.

For existing nuclear power plants the Commission specified the formulation of a systematic approach to an examination of each plant now operating or under construction for possible severe accident vulnerabilities. The systematic approach will be developed during the two years following issuance of the policy statement. The examination of each plant will be performed by the licensees. Vulnerabilities identified by this process will be evaluated against the Commission's backfit policy in deciding whether corrective actions are needed. Any generic design changes that are identified as necessary for public health and safety would be required through rulemaking.

The staff has developed an implementation program for the Policy Statement. The program will accomplish the goals of the Policy Statement within the schedule specified. The purpose of this report is to document the staff's proposed implementation program.

2. Summary of the Implementation Program

In 1982 the NRC initiated the Severe Accident Research Program (SARP) with the purpose of (1) providing a better understanding of Severe Accident phenomena, (2) developing analytical tools for the analyses of severe accidents, and (3) analyzing selected severe accident scenarios. The first phase of the program is completed and a better understanding of severe accident phenomena has been produced. Additionally, the analytical tools were developed and are being used to assess the risks associated with severe accidents. At this time six reference plants are being analyzed for their response to severe accidents and uncertainty studies are underway to assess the effects of the different sources of uncertainties on the analytical results.

Parallel with the NRC effort, IDCOR (Industry Degraded Core Rulemaking Program) on behalf of the nuclear industry has also analyzed four of the six reference plants (Peach Bottom, Grand Gulf, Sequoyah, and Zion). The IDCOR analysis and its results have already been reported(3) and presented to NRC. Based on the IDCOR presentations and on understanding gained from the severe accident research program, nineteen technical issues were identified which were treated differently by the two parties and which were judged to have a significant effect on the outcome of severe accidents.(4) NRC and IDCOR have discussed the outstanding issues, agreed on an approach to resolution and are currently pursuing resolution. A report providing most of IDCOR's contribution toward resolution of these issues has recently been published.(5)

The NRC is reviewing the IDCOR methodology and the IDCOR analyses together with the NRC calculations in order to establish an acceptable methodology and the procedural criteria for the Individual Plant Examinations.

The Severe Accident Policy Implementation Program provides for coordinated efforts to ensure the fulfillment of the policy contained in the Policy Statement. The implementation program incorporates three major elements. The first element is to formulate an integrated, systematic approach for examining each nuclear power plant now operating or under construction for possible significant risk contributors that might be plant specific and might be missed absent a systematic search. The examination will pay specific attention to containment performance in striking a balance between accident prevention and consequence mitigation. The systematic approach will include 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. From the examination, an individual plant vulnerability could be identified to be evaluated for the most cost-effective alternatives for reducing the risk significance of the vulnerability. The Commission's backfit policy will be used to decide which alternatives need to be implemented. Any generic design changes that are identified as necessary for public health and safety could be required through rulemaking. As the source term related changes get converted into rules, regulations, regulatory guides and Standard Review Plan's, IE will review the inspection program to identify and revise procedures affected by these changes.

The tasks of this element are: (1) to evaluate the methodology developed by IDCOR (Industry Degraded Core Rulemaking) and (2) to issue guidelines and criteria for the systematic safety examination for individual plants. The program is summarized in Figure 3.1 of the enclosure.

The second major element is to develop guidance on the roles of PRA's in the approval of new applications. The NRC staff will use the safety insights gained from review of past PRAs to issue guidance on the form, purpose and role that PRAs are to play in severe accident analysis and decisionmaking for future plant designs and what minimum criteria of adequacy PRAs should meet. The elements in the guidance will include (1) the combinations of deterministic requirements and probabilistic considerations appropriate as bases for severe accident decisions, (2) the definition of the minimum content of the PRAs, and (3) the criteria for the regulatory interpretations of results from the PRAS. The program element is summarized in Figure 4.1 of this plan.

The third major element is the modification of our rules, guides and other regulatory practices to reflect those changes in our scientific understanding arising from our present and continuing research effort in severe accident releases ("source terms"). While severe-accident phenomena research is still underway, the staff intends to initiate changes as soon as the availabe information warrants changes, and is proposing, herein, several such changes. The program element is summarized in Figure 5.1 of the enclosure.

The Severe Accident Policy specified not only the objectives of the Implementation Program, but also its schedule. The staff review of the IDCOR methods will be completed in October 1986. The staff will brief the Commission on the findings and recommendations for the Individual Plant Examinations including a generic letter and the guidelines and criteria by December 1986. There is no single date for the proposed changes to NRC rules and regulatory practices. Following accomplishments to date (summarized in Table 2.1), the proposed implementation schedule is summarized in Table 2.2 for the Severe Accident Policy implementation and in Table 2.3 for Source Term related changes in the rules and regulatory practices.

The Severe Accident Policy Implementation Program is described in more detail in the following sections.

Table 2.1

Summary of Accomplishments
For Severe Accident Policy Implementation
and Source Term Related Changes

IDCOR Technical Summary and Support Reports Issued	11/84
Agreement Between NRC and IDCOR On The Approach To Resolving Open Technical Issues	4/85
First Phase Of The Research Program To Upgrade The NRC's Understanding Of Severe Accidents And The Reassessment Of The Technical Bases For Estimating Source Terms	7/85
Severe Accident Policy Issued	8/85

Table 2.2

Summary of Expected Accomplishments
Severe Accident Policy Implementation

Complete the NRC Analysis Of Six Reference Plants For Severe Accidents Including Source Term Calculations	6/86
Resolve IDCOR/NRC Technical Issues	7/86
Complete the Reference Plant Sensitivity Studies (Evaluation of Uncertainties)	7/86
Complete Review Of IDCOR Methodology For Individual Plant Examinations.	10/86
Brief Commission on the findings and recommendations for the Indivdual Plant Examinations	12/86
Issues Guidance For Public Comment On The Rule Of PRAs For New Plant Applications	2/87
Issue For Public Comment Rule Changes Necessary To Resolve Generic Severe Accident Related Vulnerabilities	4/87

Table 2.3
Summary of Expected Accomplishments
Source Term Related Changes

Issue For Comment Revised SRP Section 6.5.2 Specifying The Need For Spray Additives In PWRs	9/86
Issue For Comment Regulatory Guide 1.3 And The Appropriate Section Of The SRP On Fission Product Scrubbing In Suppression Pools (BWRs)	9/86
Issue For Comment Proposed Changes To 10 CFR 50.47 and 10 CFR 50, Appendix E On Emergency Planning	2/87
Revise NRR Office Letter 16 With Respect To The Use Of Source Terms In Safety Issue Evaluation	2/87
Issue For Comment Changes In Containment Leak Rate Requirements, Including Potential Changes In 10 CFR 50 Appendix J	3/87
Revise 10 CFR 50.49 And Regulatory Guide 1.89 With Respect To The Radiation Environment For Equipment Qualification, For Comment By	6/87
Issue For Comment Revisions Of Siting Criteria (10 CFR 100) Based On New Source Term Information	10/87
Issue For Comment Revised Regulatory Guide 1.97 On Accident Monitoring And Management	12/87

3. Existing Plant Examination (Program Element 1)

The objective of this program element is to implement those portions of the Commission's Severe Accident Policy which pertain to the systematic examination of operating plants and plants with construction permits. These Individual Plant Examinations (IPEs), to be performed by the utilities themselves, are intended to identify the plant-specific vulnerabilities which contribute significantly to the overall risk from severe accidents. This program element will define an acceptable methodology for use by a utility; establish guidelines for the scope of the design and operations to be reviewed; and define the criteria for acceptability of design and operations.

The staff will review the IDCOR methods and expects to complete the evaluation in October 1986. The guidelines and criteria are expected to have been developed by October 1986 also. The staff will brief the Commission by December 1986 on the complete approval of the methods and the guidelines and criteria. The approved methods and the guidelines and criteria will be attached to a generic letter requesting the individual utilities to perform an IPE. The generic letter to licensees will outline the process for utility compliance and NRC review. Guidance will also be given for determining whether a previously performed severe accident assessment is sufficient to meet an existing plant's requirements under the severe accident policy.

3.1 Review of the IDCOR Individual Plant Examination Methodology (IPEM)

IDCOR developed a simplified method for utilities to use in performing an IPE.

Based on the risk assessment for each of the IDCOR reference plants, the IPEM consists of a detailed set of questions concerning plant design and operation. By answering these questions, the engineering staff of the utility identifies variances between its plant and the reference plant along with the effects of the variations on the severe accident vulnerabilities of the plant.

Two IDCOR methods exist: one for BWRs and one for PWRs. These methods are currently being applied by the utility owners of four PWR plants and three BWR plants to demonstrate the methods and correct any problems before submitting the methods for NRC review. The purpose of this task is to review and evaluate the IDCOR methods.

Evaluation of the IDCOR methods will be based largely on comparison with severe accident insights from three other sources. First, we will determine whether it fully covers the guidelines and criteria which we are developing for each plant type (see Section 3.2 below). These guidelines and criteria identify those aspects of plant design and operation which should be included in the IPEs. Second, the available information on the IDCOR methods is being examined to determine how well it incorporates insights from existing PRAs and the criteria from the resolutions (or proposed resolutions) of Unresolved Safety Issues (USI's) relevant to severe accidents. Finally, the IDCOR methods are being reviewed against operating experience from various plant types for completeness in the consideration of potential precursors to severe accidents.

Review of the IDCOR methods includes developing standards for an acceptable method by March 1986. The IDCOR will submit seven individual reports to

demonstrate the application of the BWR version to three plants and the application of the PWR version to four plants.

The schedule for submittal of the utility reports has not yet been decided by IDCOR. Preliminary planning for the NRC review is based on the assumption that four of the reports (two BWRs and two PWRs) will be submitted by March 1986. We expect the evaluation of each plant will take five months.

NRC review will evaluate the overall characteristics of the IDCOR methods and their application to an individual plant. First priority is given to evaluating the overall characteristics of the methods including any improvements needed for approving the methods. We will continue interaction with the ACRS during the review. We anticipate informing IDCOR of any major questions on the methods by the end of June 1986, followed by the evaluation of the methods by the end of October 1986.

3.2 Development of Guidelines and Criteria for Plant Examinations

This task will determine which aspects of plant design and operation should be examined in the IPE, and what the criteria are for acceptability. Although these guidelines and criteria will be largely deterministic in nature, they will be based on our perspective of severe accident risk. The guidelines will direct the utilities to those features of the plant which represent severe acci-

dent vulnerabilities, and those features which have a potential for significantly reducing risk. These risk perspectives will be derived from (1) analyses performed by IDCOR, (2) the reference plant integrated risk assessments being performed by the NRC-sponsored Severe Accident Research Program (SARP), and (3) previous staff experience with PRA reviews for specific plants.

Tabulated in Table 3.1 are the reference plant analyses from IDCOR and NRC which form the basis for the guidelines and criteria.

TABLE 3.1

Reference Plant Analyses Performed by IDCOR and SARP

Plant Type	Reference Plant Risk Analysis*	
	IDCOR	SARP
PWR, Large Dry	Zion	Zion
PWR, Ice Condenser	Sequoyah	Sequoyah
BWR, Mark I	Peach Bottom	Peach Bottom
BWR, Mark II	--	LaSalle
BWR, Mark III	Grand Gulf	Grand Gulf

* SARP has also analyzed the Surry plant, a PWR with subatmospheric containment. This type of plant will be included in the guidelines and criteria for large dry plants.

The task of developing guidelines and criteria will proceed in five steps: resolution of technical issues, review of the IDCOR and SARP reference plant analyses, preparation of strawman guidelines, development of proposed acceptance criteria, and definition of final guidelines and criteria.

3.2.1 Technical Issue Resolution

The NRC and IDCOR have had numerous technical exchange meetings during the past few years. The NRC and IDCOR differences were initially a larger set and have been reduced to a set of 19 technical issues.[4,5] Considerable progress has been made recently to resolve those technical issues. An important part of our overall program is completing the resolution of these technical issues. Issue resolution does not necessarily mean either that the NRC and IDCOR are in total agreement on the models or that further research is unnecessary. Instead, Issue resolution means that the sources of the differences are sufficiently understood to provide for a regulatory position. For some of the technical issues the resolution will also include definition of the range of uncertainties for the SARP sensitivity analyses. In May 1986, draft issue papers will be completed that describe the NRC position on each issue and identifying areas where further effort might be needed.

The uncertainties in the risk estimates for the reference plants stem from a variety of sources including the definition and quantification of accident sequences, and the modeling of severe accident phenomena. To derive an

overall estimate of uncertainty we will combine these individual uncertainties to obtain the overall uncertainty in mean annual risk. The first step in this process is to identify the leading sources of uncertainty and estimate the ranges over which they can reasonably vary. A task leader was assigned for these technical issue to facilitate resolutions for each plant type. The uncertainty ranges are expected to be defined by the end of May 1986 for Surry, Peach Bottom and Sequoyah. The schedule for the other three plants calls for completion of this task in June 1986.

Research on the severe accident issues will continue, and adjustments to the analyses may be required at a later date. To accommodate this possibility, we have scheduled an update. The update will (1) account for new research results that would meaningfully alter past findings which could influence Commission decisions, and (2) provide assurance that the guidance is current by reflecting the significant findings from new research. The research update is scheduled for October 1986, prior to the NRR recommendations to the Commission on the IPEs.

3.2.2 Evaluation of the Reference Plants

The purpose of this subtask is to assemble a risk profile for each type of plant, based on the available analyses specifically including those performed by IDCOR and SARP. The profiles should include event trees and core damage frequency estimates for all significant accident sequences; containment response matrices; fission product release fractions, release energy and timing; and calculations of offsite consequences. The information should be of sufficient

detail to identify the plant systems and operator actions which are potentially important to severe accident risk for each type of plant. Emphasis will be placed on both the principal contributors to risk and the plant features which are most effective in reducing risk.

The probabilistic considerations of core damage frequency estimates and containment response will guide the deletion of accident sequences considered unimportant. Because of the inherent uncertainties of probabilistic analysis low frequency sequences will be carefully considered against engineering judgment before deleting any credible accident sequence.

The schedule calls for the SARP risk profiles to be received for all the reference plants by June 1986. The reference plant evaluations are scheduled to be completed by August 1986.

3.2.3 Preparation of Strawman Guidelines

Upon completion of the risk profile for each plant, a preliminary or strawman set of guidelines will be developed. The guidelines will specify the plant features and operator actions which are considered important to ensuring acceptable risk for the reference plants. For the accident sequences which are judged to have low core damage frequency and public dose risk based on our evaluation of plant design and operating characteristics, the guidelines will specify the plant features which contribute most significantly to that result.

3.2.4 Development of Proposed Criteria

This subtask will develop acceptance criteria for the various strawman guidelines. The criteria will specify the attributes necessary to ensure acceptable performance. We anticipate a mix of deterministic and probabilistic criteria.

The strawman guidelines and proposed criteria will be complete by September, 1986.

3.2.5 Development of Final Guidelines and Criteria

The strawman guidelines and proposed criteria will be reviewed and evaluated from several perspectives. Most importantly, we will examine their applicability to other plant types in the same class. We expect to encounter cases in which an important function, performed by a particular piece of equipment at the reference plant, is handled by a different system or by an operator action at another plant. The important accident sequences at the reference plant will probably vary among plants.

The guidelines and criteria will be checked with reference to other sources of similar information. For instance, insights from existing PRA's have been compiled by a number of analysts. We will make certain that those insights are adequately covered. In addition, the guidelines and criteria will be used as a check on the IDCOR methodology (see Section 3.1). We expect that modifications to the guidelines and criteria will result from that process.

The final guidelines and criteria will be consistent with the regulatory principle for source terms (Section 5.1.2 below).

Following these reviews, the final set of guidelines and criteria will be developed for each plant type, with completion scheduled for October 1986.

3.3 Major Milestones and Schedule

The major milestones for implementing the above program are shown in this section. Table 3.2 lists the milestones for the major tasks and subtasks in chronological order, while Figure 3.1 shows the inter-relationships and dependencies of the major tasks.

TABLE 3.2

Listing of Milestones

3.1 Review of the IDCOR Individual Plant Examination Methodology

- Standards for an acceptable methodology	-	3/86
- Submittal of IDCOR reports for two BWRs and two PWRs	-	3/86
- Submittal of remaining three IDCOR reports	-	7/86
- Report to IDCOR on major shortcomings of their methodology	-	7/86
- Evaluation of the application of the IDCOR Methodology to seven plants	-	10/86

3.2 Development of Guidelines and Criteria for Plant Examinations

3.2.1 Technical Issue Resolution

- Define uncertainty ranges for Surry, Peach Bottom & Sequoyah	-	2/86
- Define uncertainty ranges for Zion, Grand Gulf & LaSalle	-	3/86
- Draft NRC/IDCOR issue papers	-	5/86
- Final NRC/IDCOR issue papers	-	7/86

3.2.2 Evaluation of Reference Plants

- Completion of IDCOR severe accident evaluations (with uncertainty analysis) (4 or 5 plants completed in March'86)	-	7/86
- Completion of SARRP Risk evaluations:		
Surry	-	4/86
Peach Bottom	-	5/86
Zion	-	5/86
Sequoyah	-	6/86
Grand Gulf	-	6/86
- Complete Reference Plant Risk Profile	-	8/86
- Evaluation of Reference Plants	-	8/86

3.2.3 Preparation of Strawman Guidelines

3.2.4 Development of Proposed Criteria

- Strawman Guidelines & Proposed Criteria
  - Peach Bottom - 6/86
  - Other Plants - 9/86

3.2.5 Development of Final Guidelines & Criteria

- Final Guidelines & Criteria - 10/86

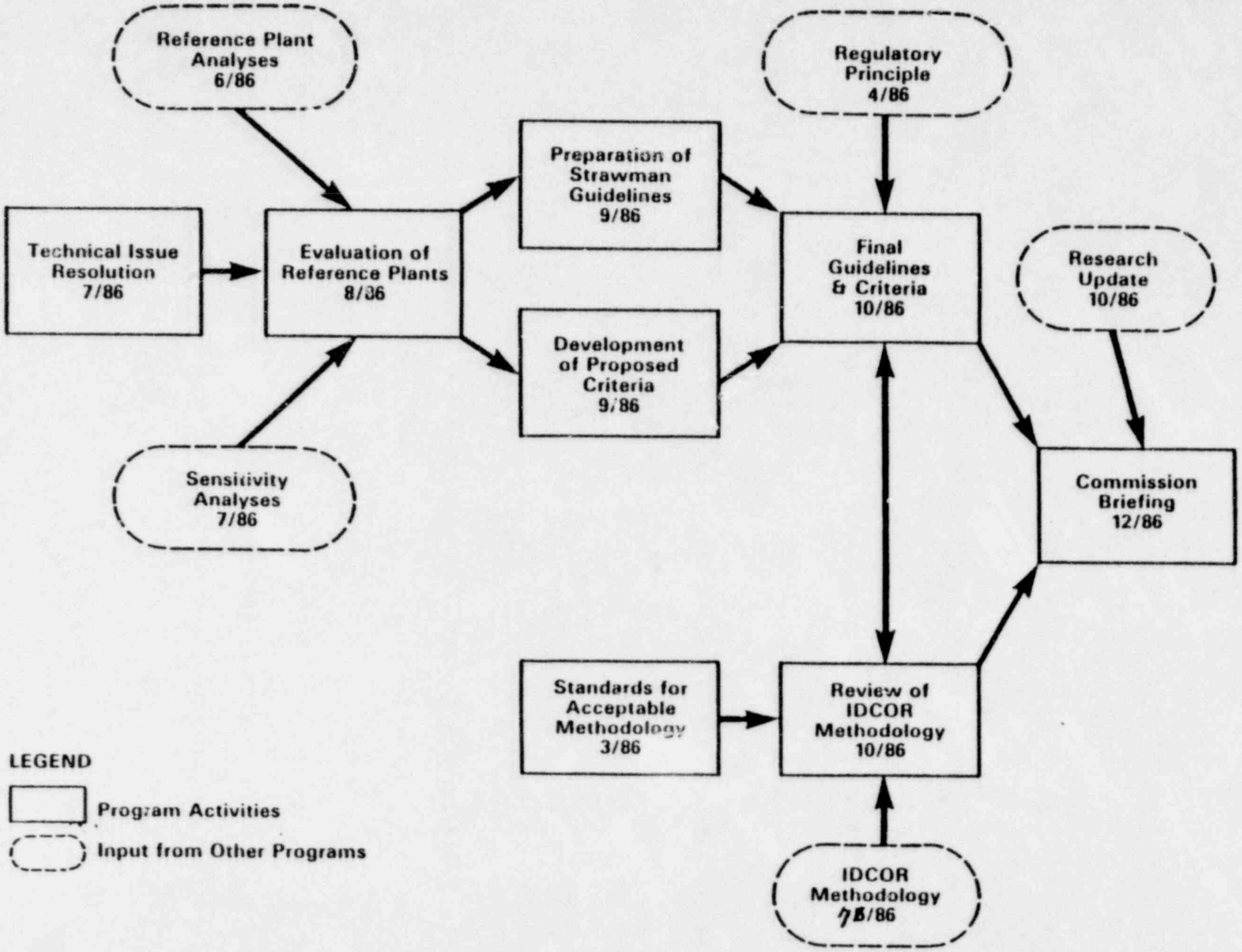

Figure 3.1
Program Element 1 - Development of Guidance for Individual Plant Examinations

4. Development of Guidance on the Role of PRAs (Program Element 2)

For new applications the Severe Accident Policy requirements include:

"c. Completion of a PRA and considerations 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 judgement complemented by PRA."

The NRC staff plans to develop guidance on the role of PRAs for new applications that utilizes the safety beneficial features of past PRAs and PRA-type analyses. While performing past PRAs, a large number of potential hazards were integrated into risk perspectives to discriminate what was of central importance to reactor safety from what was not of central importance. Although large uncertainties are associated with PRA numerical assessments, potential safety improvements were identified by the systematic performance of PRA and PRA-type analyses. Also, past PRAs have enhanced communications among design, operations, and maintenance groups with differing responsibilities or areas of expertise.

The past regulatory approach to approving applications has treated uncertainties by adopting the philosophy of "defense in depth and conservative

judgment." The degree of conservatism (the size of each safety margin) was not necessarily uniform throughout the design of the plant. Questions remained concerning both the magnitude of the overall safety margins and the potential for inconsistent requirements. Currently, PRAs have been providing overall risk profiles (identifying prominent risk contributors) and have determined how the plant's components could interact to cause undesirable safety consequences. PRAs have facilitated decisions that balanced regulatory attention between better containment performance and core-damage prevention. Some leading contributors to risks have been reduced by plant modifications. The leading sources of uncertainties have become the subject of further study and research. All these benefits accrued within the current regulatory process. Thus, the objective is to define those roles for PRAs that best provide for an independent perspective on new designs to complement the current regulatory requirements. The objective is not to substitute another set of regulatory requirements.

The Guidance on the Role of PRAs for future applications will be developed by the following three tasks: (1) to establish the combinations of deterministic requirements and probabilistic considerations that form the bases for decisions on severe accidents, (2) to describe the minimum acceptable content of PRAs on future plants, and (3) to define the criteria for the regulatory review and interpretation of the PRA results. The three tasks in this objective will rely upon the collective experience from prior PRA reviews and PRA related programs.

4.1 Deterministic Requirements

Since the Severe Accident Policy stresses deterministic engineering analysis and judgment, the first task is to establish the combinations of deterministic requirements and probabilistic considerations that future applications must provide to assure public health and safety against severe accidents. The deterministic requirements for severe accidents should neither duplicate nor replace the current requirements for Design Basis Events as documented in Section 15, Standard Review Plant, NUREG-0800. Rather the deterministic requirements established in this task will provide the minimum deterministic requirements to provide reasonable assurance of public health and safety against severe accidents.

The staff has recommended protective measures against severe accidents based upon reviews of PRAs on specific plants, e.g., Indian Point[6], Limerick[7], and GESSAR[8]. Additionally, some licensees have voluntarily increased safety margins at their plants based upon the results from specific PRAs. Using this experience derived from past PRAs and their insights, the staff will review the active safety issues for relevance to the severe accident issue. The safety issues to be reviewed will include the Unresolved Safety Issues[9] the High and Medium Priority Generic Safety Issues[9] , and any issues resulting from the Individual Plant Examinations of the six reference plants (see Section 3). Using engineering analyses supplemented by PRA insights, the staff will select from the active safety issues those deterministic criteria that must be considered specifically for severe accidents. The acceptable combinations of probabilistic considerations and deterministic requirements will be established.

4.2 Acceptable Content of PRAs

Since the Severe Accident Policy requires completion of a PRA, the second task is to describe the minimum acceptable content of PRAs. The minimum acceptable content will be those elements that are essential for the intended use of a PRA. By subtask 4.2 the staff will define the acceptable structure of a PRA or a PRA-type assessement. The staff will also define the products of such assessments that are expected to be material in regulatory decisions. The staff guidance will attempt to allow for alternative PRA-type methodology provided the essential elements of the minimum acceptable content of PRAs is present.

The elements that are essential will be described, such as: the scope of the hazards (discussed further in Section 7), the applicability of available data, the use of individual plant data, the treatment of operator error rates, the assessment of the effects of common-mode failures, the search for common-cause failures, the extent of modeling the support systems dependencies, the format of the PRA reports, the identification of the sources of uncertainties, and the assessment of the magnitude of the included uncertainties.

To accomplish subtask 4.2, the staff will review and evaluate the Draft Probabilistic Safety Analysis Procedures Guide[10]. The draft was prepared for another program and will be reviewed to assure that it accommodates the Severe Accident

Policy. The NRR staff will review the methods used in SARP for the integrated assessment of the reference plants.[11] The NRR staff will review the Interim Reliability Evaluation Program Procedures Guide[12] and the Simplified IDCOR Methodology (see Section 3) for completeness in describing the essential elements of a PRA, e.g., the extent of modeling of the support systems and the level of detail to be included. The staff does not expect to prescribe PRA methods rather the staff intents to identify the minimum content of the PRA.

The draft of the Probabilistic Safety Analysis Procedures Guide[10] does not treat the assessment of containment performance and calculated fatalities. Part of subtask 4.2 will be to supplement the Guide by the development of the minimum acceptable content of a PRA for containment performance and calculated acute and latent fatalities. Since draft guidance for containment response and calculated fatalities has not yet been developed, this effort is expected to compose the larger part of the subtask.

4.3 Criteria for the Regulatory Review and Interpretation of the PRA Results

Because the Severe Accident Policy requires the consideration of the vulnerabilities that PRAs expose, the third task is to define the criteria for the regulatory review and interpretation of the PRA results. For this subtask the staff will define how the results of the PRA will be systematically folded into the regulatory process. Specific aspects of the process will be defined for accomplishing considerations such as the following: (1) the use of the PRA results in the Environmental Statements, (2) the ranking of contri-

butors by their importance to risk, (3) the establishing of thresholds to trigger responses to potential safety problems, (4) the monitoring of the assessed risks during operations and maintenance, and (5) the evaluation of requested changes to the license conditions. Plant specific PRAs have supplemented some regulatory decisions in the recent past. [6,7,8, & 13] The past interpretations that were deemed appropriate for plant-specific decisions will be evaluated as part of this task.

The state of the PRA art is such that there are a diversity of PRA methods. The review criteria for PRAs on future-applications will be defined as part of this subtask. The review criteria will define how the PRA will be reviewed for (1) accuracy against the plant's design and operation, (2) consistency with prior assessments which consisted of similar elements, and (3) clarity of the results. The PRA review criteria would yield a means of grading the PRA. Also, subtask 4.3 will address the limitations that the quality of the PRA places on the interpretations justified by the PRA. To accomplish subtask 4.3, the staff will review and evaluate the draft Probabilistic Safety Assessment Review Manual[14]. The draft was prepared for another program and will be reviewed to assure that it accommodates the Severe Accident Policy. In addition, the staff will rely upon experience accrued during the review and evaluation of past PRAs on specific plants.

Subtask 4.3 includes clarifying the use of the safety goals for interpreting the PRA results. The safety goals are expected to be of value in the review of standard plant designs and in considerations of exemption requests. The goals would be taken into account as one factor among others in reaching regulatory decisions but would not constitute sharp thresholds for acceptance or non-acceptance without consideration of other factors. (See Section 6.2.1 for further discussion of the Safety Goals).

A recognized constraint in the use of PRA results for regulatory decisions has been the magnitudes of the uncertainties associated with the numerical results from PRAs. Subtask 4.3 includes clarifying the appropriate weight to be given the uncertainties when interpreting the PRA results on a future plant. The experience gained from the sensitivity studies on the reference plants (see Section 3) will be used to identify which uncertainties must be considered for future plants. Since the guidance on the role of PRAs precedes the guidance for the Individual Plant Examinations, the clarification of uncertainties considerations coming from Subtask 4.3 will also be useful during the evaluation of the results from the Individual Plant Examinations.

Consistent with the Severe Accident Policy objective "to ensure that the recognized level of safety is commensurate with estimates based on safety analyses used in regulatory decisions", the staff will develop criteria for the incremental changes in the assessed risks. The criteria will be developed for incremental risks due to changes in either the plant's configuration or relevant events at other facilities. Value-impact concerns will be a major consideration for proposed criterion that could lead to corrective action at a plant.

4.4. Major Milestones and Schedule

The major milestones for implementing the above program are shown in this section. Table 4.1 lists the milestones for the major tasks and subtasks in chronological order, while Figure 4.1 shows the inter-relationships and dependencies of the major tasks.

Table 4.1

Listing of Milestones
Development of Guidance on the Role of PRAs

4.1	Draft Deterministic Requirements	-	12/86
	- Deterministic assessments of reference plants	-	8/86
	- Evaluation Report on Existing and Needed Deterministic Requirements	-	10/86
4.2	Draft Guidance on Minimum Content of PRAs	-	12/86
	- Reference plant analysis	-	8/86
	- Evaluation of IDCOR method for IPES	-	10/86
	- Draft Guidelines & Criteria for IPEs	-	10/86
	- Draft procedures for Core Damage Frequency Assessment	-	10/86
	- Draft procedures for containment and consequence analysis	-	11/86
4.3	Draft Criteria for Regulatory Use of PRAs	-	12/86
	- Revised Safety Goal	-	6/86
	Containment Performance Objective	-	6/86
	- Reference Plant Sensitivity Studies	-	8/86
	- Reference Plant Analysis	-	8/86
	- Clarification of the use of the safety goal in view of uncertainties	-	9/86
	- Draft criteria for assessing incremental changes in risks	-	9/86
4.5	Commission Paper on Guidance on the Role of PRAs	-	1/87
	Commission Approval to Issue Guidance for Comment		2/87

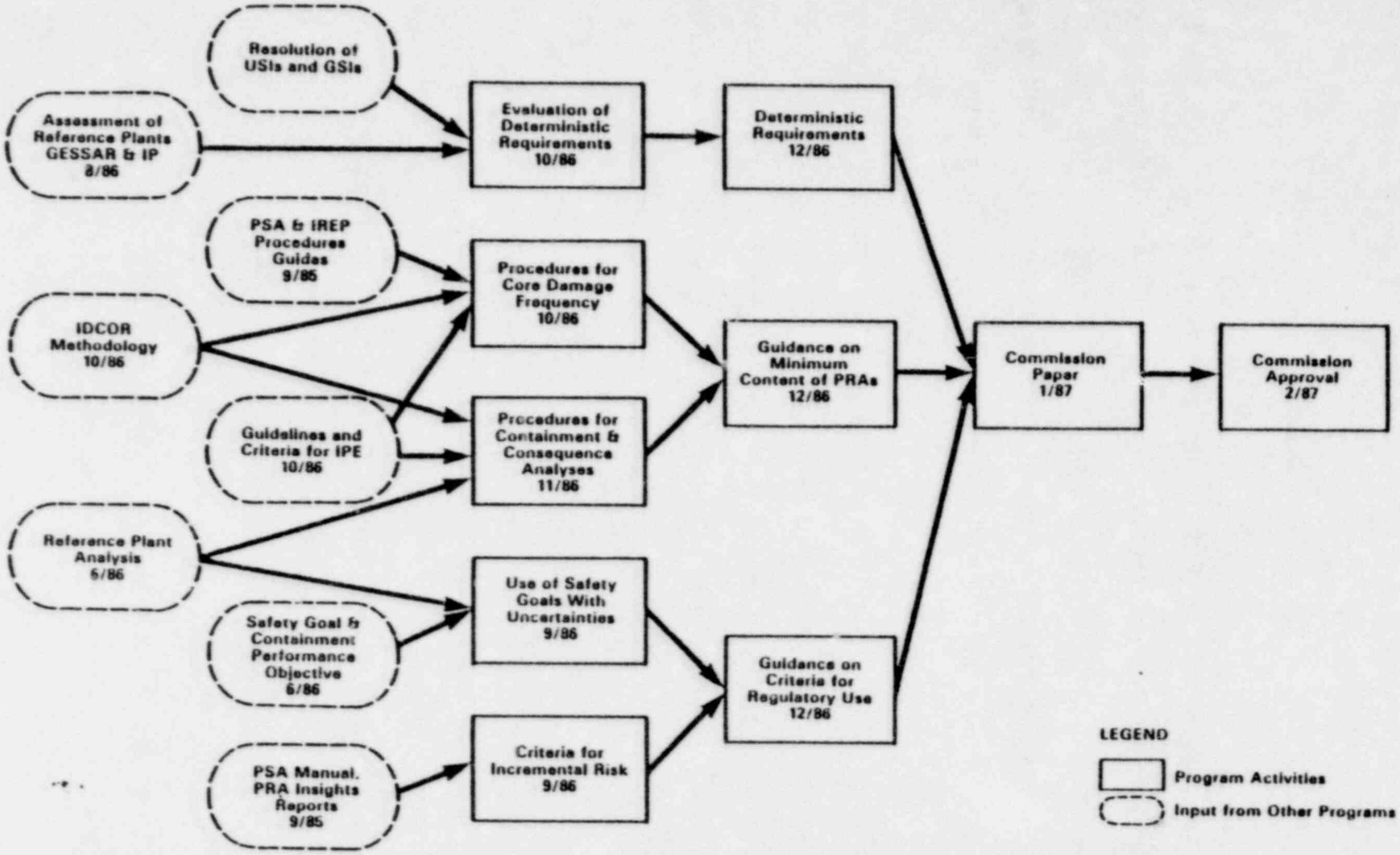

Figure 4.1 Program Element 2 - Development of Guidance on the Role of PRAs

5. Changes in Rules and Regulatory Practice (Program Element 3)

In carrying out the Severe Accident Policy Implementation Program, the staff expects to propose a number of changes to NRC rules as well as other changes in regulatory practice. These changes could arise from research (both published and on-going) regarding radioactivity releases under severe accident conditions ("source terms") as well as other insights expected to be gained through the evaluation of severe accidents, in general. The specifics of the implementation plan that follows, groups the changes into these two broad areas and provides an itemization as well as a preliminary schedule. In addition to these sources of possible changes, it should be noted that any insights arising from individual plant examinations noted in Section 2.1 that are considered to have generic applicability, will also be considered as candidates for changes.

5.1 Source Term Related Changes

A number of changes in rules and regulatory practices are expected from our improved understanding arising from the extensive research efforts on radioactivity releases under severe accident conditions ("source terms"). The implementation of such changes requires (1) a capability to perform source term calculations, (2) selection of a regulatory principle, or framework, in connection with evaluation of plants beyond the current design basis, (3) development of new forms of source terms, and (4) revision of the

affected rules and other regulatory practices. The sections that follow discuss these in greater detail.

It is also important to note that changes in source term estimates are not expected to treat two areas: (1) the adequacy of on-site property damage insurance and (2) the perceived need for offsite indemnification requirements (Price-Anderson).

5.1.1 Establish Capability for Source Term Calculations

Efforts to initiate source term related changes in rules and regulatory practices will require the development of new source terms. This, in turn, will require a capability to perform source term calculations for both design basis as well as severe accidents. Source term calculations can be performed by the Source Term Code Package[15] or the MELCOR code[16]. A capability to use the Source Term Code Package (STCP) to perform source term calculations will be established with at least two national laboratories. This capability already exists at one laboratory since they are the principal contractor for RES in this area as well as the authors of the STCP.

The new source terms will be generic and cover at least a group of plants, for example, BWR's with Mark I containments. Variations in the source terms due to any plant specific design differences will be evaluated. The available industry assessments will be considered also, e.g., IDCOR calcuations using MAPP.

5.1.2 Selection of Regulatory Principle

The current regulatory framework, involving the use of the TID-14844(17) in-containment release assumptions in connection with the evaluation of design basis accidents, treats design basis events in an overly conservative manner with respect to source terms, but may be non-conservative with regard to the effects of severe accidents. An important goal is the development of a revised regulatory framework or principle that employs the use of consistent, realistic source terms. Such a regulatory principle will specify which severe accidents need to be considered for calculations of source terms. It will establish the logic and bases of limiting source term calculations to these selected sequences, and it will define the selected sequences.

The next two sub-tasks (development of new source terms and revision to the regulations) depend to a large extent upon the selection of the regulatory principle. Timely selection of the regulatory principle (probably by April, 1986) is essential to the progress of these sub-tasks. Potential options will be drafted by the end of February, 1986. Technical Assistance (probably from BNL) will be utilized to provide assistance in examining severe accident sequences and their associated source terms. A meeting with industry representatives (AIF) and an ACRS meeting will be scheduled shortly after. Selection of the regulatory principle will follow. A Commission paper informing the Commission of the selection will be forwarded by May 1986.

5.1.3 Development of New Forms of Source Terms

Given the new regulatory principle, source terms will be calculated for the six reference plants. The expectation is that many of the needed calculations will be available from the reference plant analyses presently being performed. Any additional calculations will be run as needed. Based on the reference plant calculations, acceptable input, assumptions and methodology will be specified for future source term calculations, including an acceptable treatment of the uncertainties associated with such calculations.

Differing applications of source terms are expected to require different forms. The various uses (see next sub-task) will be reviewed to determine how many forms of source terms are needed. One acceptable form will be detailed calculations to provide source terms as it is to be done for the reference plants. In addition to this, it is likely that a need exists for a simplified source term form to predict release from the containment. A third form of source term is needed to facilitate the qualification of safety-related equipment located inside containment.

With regard to a simplified source term for release from containment, the goal is to develop tables or simple procedures applicable to a plant type. Based on the reference plant calculations, plants will be grouped into plant types for this purpose. Major design variations as well as variations in operating procedures that are expected to influence this source term will be

identified within each plant type. Source term sensitivity studies will be performed in order to bound the effects of these variations. The results will be arranged either in tables or will be given as functions of selected input variables. An appropriate margin to cover uncertainties will be included explicitly.

Equipment qualification source terms will be calculated at various locations inside containment for the reference plants. Based on the reference plant results, generic source terms by plant type could be developed for equipment qualification.

5.1.4 Revision of Rules, Regulatory Guides and the Standard Review Plan

A major piece of information needed to initiate revision of rules, regulatory guides and the Standard Review Plan (SRP) is the severe accident analyses and the resulting risk profile of the six reference plants.

Source term analyses for reference plants are expected to become available in early 1986 with completion of the source term analyses expected in June 1986. While many changes in NRC rules and regulatory practices must await the insights to be gained from the reference plant analyses, the staff intends to initiate changes as soon as the available information warrants such changes.

The potential changes have been grouped into those that are anticipated to be initiated prior to availability of the reference plant analyses (short-term

changes); those where a background effort is expected to proceed in parallel with the reference plant analyses with rulemaking or other changes commencing after completion of the reference plant analyses (intermediate term changes); and those where effort will commence only after completion of the reference plant analyses (long-term changes). The individual items are discussed below.

Short Term Changes

Three short-term changes are anticipated.

(1) Revised treatment of severe accidents in near-term Environmental Impact Statements (EIS)

The South Texas draft EIS presented a discussion of risks using both WASH-1400[18] source terms as well as those using the insights of BMI-2104[15] and NUREG-0956.[19] The results show little sensitivities to using these two groups of source terms since the South Texas site is not highly populated. The major difference is that no early fatalities are predicted with the newer set of source terms at the South Texas site.

Since the Commission's policy statement on "Nuclear Power Plant Accident Considerations under NEPA" (45 FR 40101, June 13, 1980) which forms the interim guidance for the staff's treatment of severe accidents in EIS's, already directs the staff to discuss health and safety risks "in a manner that fairly reflects the current state of knowledge", the staff expects no further changes in rules or regulatory practice resulting from the use of revised source term information in EIS's.

(2) Removal of spray additives in PWR's.

Source term research results for severe accidents (NUREG-0772 and NUREG-0956) have indicated that iodine fission products released into containment are not predominantly in the form of elemental iodine.

Chemical additives such as sodium hydroxide (NaOH) are usually added to PWR spray systems to enhance the removal of elemental iodine. These add complexity and also represent a potentially corrosive, damaging environment in the event of inadvertent spray operation. The regulatory requirements for spray-additive systems are based on the assumption that the elemental iodine will be immediately releasesed into the containment. However, the source term research results indicate that the time of any iodine release will vary. Further, some calculations and experiments indicate that the iodine may be predominantly in the form of cesium iodine. However, some recent experiments indicate that volatile forms of iodine may still be present.

For these reasons, the staff will study the criteria for the use of spray

these reasons, the staff will study the criteria for the use of spray

additives for PWR's. Some form of post-accident pH control may still be necessary to maintain an appropriate degree of alkalinity in the containment sump solution. The pH control will prevent evolution of dissolved iodine and provide for long term equipment survivability. The staff will be reviewing the available information during the next months. Anticipated changes concerning spray additives will not involve a revision of any rule or regulatory guide, but will require a revision of SPR Section 6.5.2.

(3) Credit for fission product scrubbing in suppression pools (BWRs).

Recent research results (NUREG-0956) also indicate that BWR suppression pools can scrub out fission products (other than noble gases) released into them. Present staff practice gives no credit for such scrubbing under postulated design basis accident conditions. NRR will examine, together with RES, the available data on fission product scrubbing and will make an appropriate determination of the degree of credit to be given for fission product scrubbing. The possibility of sequences that bypass the suppression pool will also be taken into consideration. Implementation of this position is expected to involve a Generic Letter and/or a change to the Standard Review Plan (SRP), a revision to Regulatory Guide 1.3, and consideration of 10 CFR 100.

Intermediate Term Changes

Five areas have been identified where a background effort is expected to proceed in parallel with the six reference plant study. These areas are discussed briefly.

(1) Emergency Planning

Possible changes in emergency planning requirements including changes in the sizes of the plume emergency planning zone and the implementation of a phased or graded response will await completion of the reference plant analyses. Source term analyses are expected to be completed by June 1986. The staff will then use the insights gained to develop options on proposed modifications in emergency planning requirements by November 1986.

(2) Containment Leak Rates

As a result of source term changes, it is anticipated that somewhat higher allowable containment leak rates will be found acceptable, while at the same time the staff intends to propose requirements intended to provide assurance against an undetected breach of containment integrity. Staff studies are expected to begin in April 1986 with the availability of the source term analyses. This may involve changes to 10 CFR 50 Appendix J.

(3) Control Room Habitability

Control room leak-tightness and air filtration requirements are largely determined by iodine concentrations postulated to be released in a design basis accident. The staff will begin a re-assessment of this area

beginning in April 1986. This may result in changes in Regulatory Guide 1.52 as well as SRP Section 6.4.

(4) Environmental Qualification of Equipment

Present safety-grade equipment is qualified for the radiation environment defined by the TID 14844 assumptions. As part of its current FY 1986 plan, the Office of Research is planning to perform comparisons of the radiation environment given by the TID-14844 release assumptions, and those given by the severe accident scenarios calculated in BMI-2104.

Commencing with the availability of information from the refernece plant analyses, the staff will re-assess the radiation environment that equipment should be qualified to. This may result in changes to 10 CFR 50.49 and Regulatory Guide 1.89.

(5) Safety Issue Evaluation

Prioritization of safety issues is made using WASH-1400 accident source terms. The staff anticipates revising the source terms used in prioritization to incorporate ~~with~~ the insights gained from source term research. Because the relative importance of the prioritized issues is not expected to change, no reprioritization is needed at this time. Revision of the methodology is expected to commence in May 1986, and to be completed by November 1986.

The use of WASH-1400 source terms are referenced in NUREG/BR-0058 ("Regulatory Analysis Guidelines of the U.S.N.R.C."). Since guidance to the staff in this area is via management direction through NRR Office Letter 16 (Regulatory Analysis Guidelines), which reference NUREG/BR-0058, no regulations, Regulatory Guides or SRP sections need be changed.

Long Term Changes

Two areas have been identified where potential changes in rules and regulatory practice are anticipated to involve long-term efforts. These areas are listed below.

(1) Siting

Revision of siting criteria (10 CFR 100) to incorporate the insights gained from new source term research is expected to require a moderate effort over the next one to two years.

(2) Accident Monitoring and Management

Experience gained from the analyses and assessments of the six reference plants will be used to develop generic guidance on accident monitoring and management. Reassessment of instrumentation to monitor accident conditions (Reg. Guide 1.97) could commence at the end of 1986.

5.2 Severe Accident Related Changes

Several changes in rules and regulatory practice are expected to arise from developments other than source terms. Severe accident related changes include the development of containment performance criteria, if needed, and the resolution of generic vulnerabilities arising from an improved understanding of severe accidents. The implementation for these items is discussed below. It should be noted that these items do not include Generic Safety Issues (GSI) and Unresolved Safety Issues (USI) identified and resolved by other means, but address only potential changes arising from severe accident issues.

5.2.1 Development of Containment Performance Criteria

Most existing containment requirements, like current regulations, are based upon design basis accidents. Examples are containment design requirements, such as pressure and temperature, as well as containment isolation and cooling requirements. A notable exception is the containment leak rate requirement, which combines design basis accident conditions (pressure and temperature) with a postulated radiological accident assumption (from TID-14844) which is essentially a severe accident condition.

Since the main purpose of containment is the retention of fission products in the event of a major accident, the question arises whether containments could accommodate certain severe accident conditions.

Somewhat independently, as part of the safety goal development, an effort is underway to develop a containment performance design objective. This design objective will express the Commission's desire with respect to containment performance, but will not set any specific enforceable requirement for containments.

The purpose of this task is: (1) to review current containment requirements in view of the six reference plant assessments and the proposed containment performance design objective; (2) to decide whether there is a need for containment performance criteria; (3) to develop containment performance criteria, if such a need exists; and (4) to incorporate the criteria in the regulations through rulemaking.

Both the assessment of the six reference plants and development of a draft containment design objective are scheduled to be completed by June 1986. The need for containment performance criteria will be assessed by September, 1986. If the need exists, options for containment performance criteria will be formulated by October 1986. Recommendations for proposed criteria will be prepared for Commission consideration by February 1987.

5.2.2 Resolution of Generic Vulnerabilities

In accordance with the Severe Accident Policy Statement enunciated in NUREG-1070, the staff will investigate any generic vulnerabilities and propose resolutions for those identified. Based upon the findings from the six

reference plants which are expected to become available starting about April 1986, the staff, in conjunction with other efforts that include the resolution of Unresolved Safety Issues (USI) and Generic Safety Issues (GSI), will propose appropriate changes to correct any generic vulnerabilities identified. Identification of generic vulnerabilities will be made by September 1986, upon the completion of the reference plant study. A proposed resolution for each of these will be prepared by the end of 1986. A Commission paper will be prepared which identifies those items which would involve significant potential changes to rules and regulatory practice. The Commission paper will be forwarded by April 1987.

5.3 Major Milestones and Schedule

The major milestones for implementing the above program are shown in this section. Table 5.3 lists the milestones for the major tasks in chronological order, while Figure 5.3 shows the inter-relationships and dependencies of the major tasks.

TABLE 5.3

LISTING OF MILESTONES

5.1.1 Establish Capability for Source Term Calculations

- Source Term Code Package (STCP) operational at BNL and BCL	- 3/86

5.1.2 Selection of Regulatory Principle

- Draft potential options	- 2/86
- Meet with industry & ACRS	- 3/86
- Selection of regulatory principle	- 4/86
- Forward Commission paper on selection	- 5/86

5.1.3 Development of New Forms of Source Terms

- Calculate source terms for 6 reference plants	- 6/86
- Evaluate uncertainty of source term calculations	- 8/86
- Propose new forms of source terms	- 9/86
- Meet with industry and ACRS	- 10/86
- Development of new source terms	- 12/86

TABLE 5.3

LISTING OF MILESTONES (Cont'd)

5.1.4 Source Term Related Changes

- Initiate work on short-term changes	- 1/86
- Meet with industry and ACRS	- 2/86
- Prepare revisions to SRP's and R.G.'s	- 5/86
- Issue SRP's and R.G.s for comment	- 9/86
- Initiate work on intermediate changes	- 5/86
- Calculate source terms for 6 reference plants	- 6/86
- Meet with industry and ACRS	- 7/86
- Prepare revisions to rules, R.G.'s and SRP's	- 11/86
- Issue revised rules, R.G.'s and SRP's for comment	- 2/87 to 6/87
- Initiate work on long-term changes	- 10/86
- Prepare revisions to rules, R.G.'s & SRP's	- 8/87
- Issue revised rules, R.G.'s SRP"s for comment	- 12/87

5.2 Severe Accident Related Changes

5.2.1 Development of containment Performance Criteria

- Complete containment assessment of 6 reference plants	- 6/86

TABLE 5.3

LISTING OF MILESTONES (Cont'd)

- Develop containment performance design objective	- 6/86
- Assess need for containment performance criteria	- 9/86
- Develop & Select options for containment performance criteria	- 10/86
- Present recommendations to Commission	- 2/87

5.2.2 Resolution of Generic Vulnerabilities

- Complete assessment of 6 reference plants	- 8/86
- Identify generic vulnerabilities	- 9/86
- Propose resolutions	- 12/86
- Forward Commission Paper	- 4/87

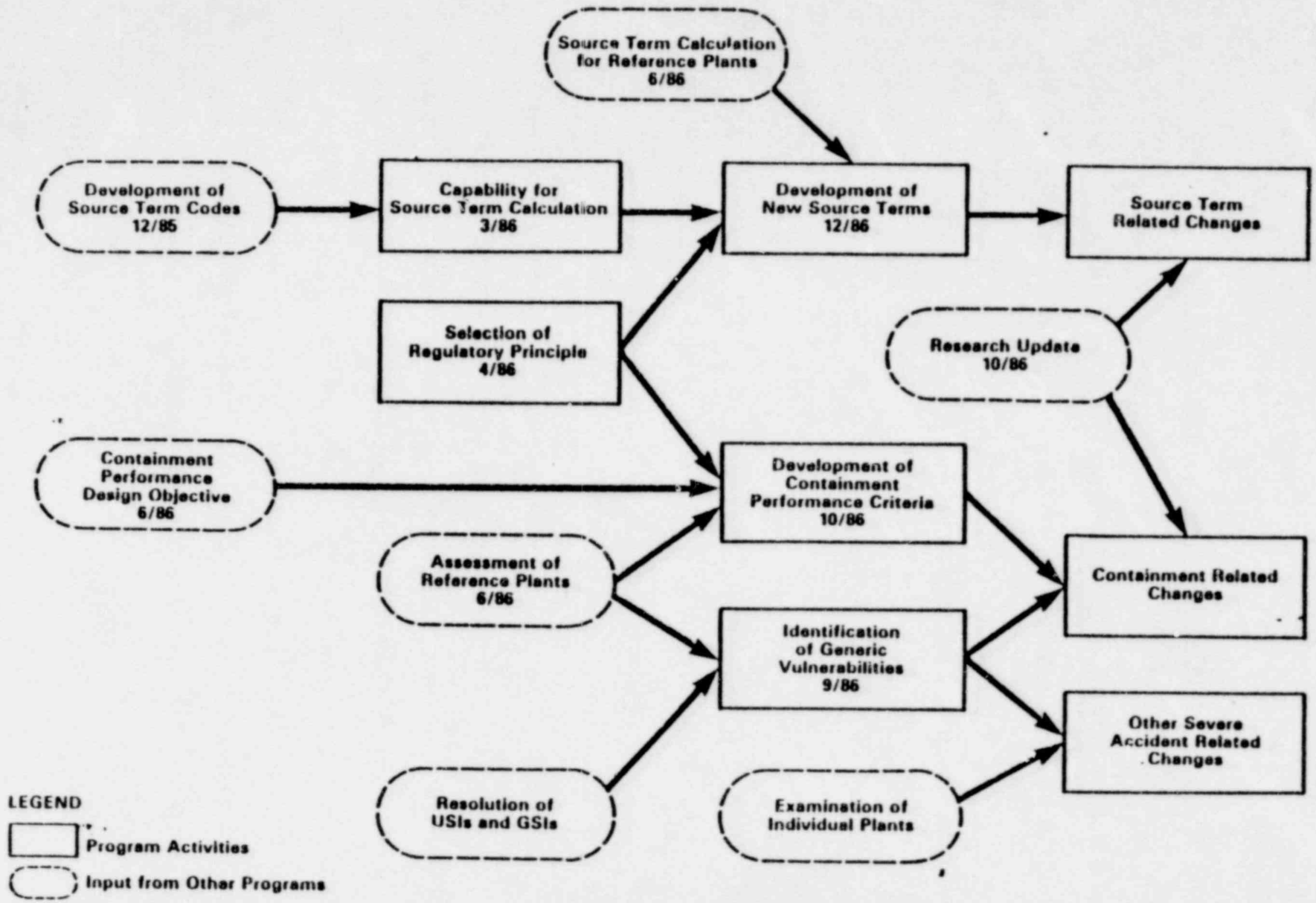

Figure 5.1 Program Element 3 - Changes in Rules and Regulatory Practice

6. Interdependence and Relationships With Other Programs

The implementation program milestones will use both the culminated studies from past efforts and future collegial efforts by the NRC staff (RES and NRR) and industry groups (e.g., IDCOR). The interfaces among these efforts are important for completing the program on schedule.

6.1 RES Programs

RES programs have extensively studied key questions within the severe accident issue. The major products from the RES studies and their milestones that will be used in the implementation program are tabulated below.

The Reassessment of the Technical Bases for Estimating Source Terms (Draft NUREG-0956)[19] provides the culmination of several years of work focused on the science and engineering of the phenomena expected to occur within a spectrum of core damage accidents. The Nuclear Power Plant Risks and Regulatory Applications Report (NUREG-1150)[20] will provide the results from the application of the current collegial understanding about severe accidents to the analyses of the six reference plants. Those experts most closely associated with the research work performed the application. In addition to other uses for NUREG-0956 and NUREG-1150, NRR will use the reference plant calculations as they become available (1) to establish the Individual Plant Examination Guidelines and Criteria and (2) to develop new source terms in support of proposed changes in the current regulatory practices. These uses were described in Sections 3 and 5.

TABLE 6.1

RES Major Products and Milestones
for Use in the Implementation Program

Item	Product	Milestone
1.	Revision to NUREG-0900 Supplement on Research Plan for Severe Accidents in LWR's	6/86
2.	Complete the NRC Analyses of Six Reference Plants for Severe Accidents, including Source Term Calculations	6/86
3.	NUREG-0956, Reassessment of the Technical Bases for Estimating Source Terms - Final	7/86
4.	Plant Sensitivity Studies	7/86
5.	NUREG-1150, Reference Plant Assessment - For Comment	8/86
6.	Status Report on Update of Severe Accident Research.	10/86

6.2 NRR Programs

Historically, rules and regulatory practices were based on Design Basis Accidents. Nevertheless, the NRC addressed severe accident issues in various forms, e.g., the development of safety goals, the resolutions of USIs and GSIs, the review of PRAs, and the hydrogen rule. The means whereby the implementation program will use the knowledge gained from these efforts are described below.

6.2.1 Safety Goals

The proposed NRC safety goals, currently in final stages of staff evaluation at Commission direction, include a decision matrix concerning mortality risks and core-melt frequency and a benefit-cost guideline for safety-cost tradeoffs. A containment performance design objective is currently being developed for possible later addition. The goals, if promulgated by the Commission, are expected to aid in developing and reviewing safety regulations and generic regulatory practices. They would also be of value in the review of standard plant designs and in consideration of exemption requests and of possible backfits. The goals would be taken into account as one factor among others in reaching regulatory decisions and would not constitute sharp thresholds for acceptance or non-acceptance.

The decision matrix--would be available as a partial standard for evaluation of the significance of the findings with respect to the six reference plants and

with respect to other individual plants examined. In these evaluations, the quantitative comparisons of plant-review findings against the matrix may consider what fraction of a risk could be reasonably allocated to a particular potential accident cause or scenario. They would also need to take into account the uncertainties in the PRA estimates, the sensitivity of the comparison to assumptions concerning poorly understood phenomena or quantities, and the potential safety consequences of such uncertainties and sensitivities. Regulatory actions for existing plants would reflect consideration of such quantitative probabilistic comparisons of estimated levels of risk along with other factors in engineering analysis. Unless risk levels estimated to be present are by themselves unacceptably high, regulatory decisions would, in addition to estimated risk levels, take into account the relation between the risk reduction achievable by a backfit and the total net cost (to the licensees, the government, and the public) of making the backfit. The benefit-cost guideline of the matrix would provide a quantitative trade-off standard to aid in making such safety benefit vs. cost evaluations.

The core-melt and mortality risks and the benefit-cost guidelines would be particularly useful in evaluations involving outliers--where a feature evaluated is either clearly too risky and clearly cost-beneficial to fix or clearly not so. When the comparison of the findings with the matrix is equivocal, i.e., when the uncertainties are large in comparison with the differences from the quantitative objectives and benefit-cost guidelines, other factors would have to guide decision.

6.2.2. Unresolved and Generic Safety Issues

The implementation of the Severe Accident Policy Statement calls for the technical resolutions of all applicable USIs and all applicable high/medium priority GSIs identified in NUREG-0933[9] on future applications. Technical resolutions are accomplished through engineering evaluations and the incorporation of the safety issue within the individual plant's risk assessment. The technical resolutions should demonstrate that either (1) each USI and GSI is not applicable to the specific design, (2) the PRA risk profiles show the risk from the USI or GSI to be an insignificant contributor, or (3) design improvements have been shown to reduce the risks from the USI or GSI to a tolerable level, i.e., compared with the proposed Safety Goals. Where necessary, any limitation in the PRA methods to technically resolve the safety issue would be overcome by regulatory judgment.

Some of the safety issues that were resolved in the past are related to the severe accident issue. Significant industry and staff resources were required to bringing about the programmatic resolutions, e.g., the ATWS issue which took a decade to resolve. Because the implementation program will provide for the minimum additional requirements uniquely to protect the public against severe accidents, the implementation program assumes that all requirements developed to formally resolve past safety issues will be in place at the plants.

The ongoing programs to resolve presently unresolved safety issues will be closely coordinated with the Severe Accident Policy implementation program. The coordination will assure that any requirements developed to resolve a safety issue and related to severe accidents will not conflict with the minimum requirements developed from this implementation program. The staff will examine all ongoing safety issues to determine which issues hold the potential to develop requirements related to severe accidents. From preliminary considerations, most of the USIs could develop requirements related to severe accidents. Two of the most likely USIs are A-44, Station Blackout, and A-45, Shutdown Decay Heat Removal Requirements.

The staff will endeavor to profit from the knowledge gained on both past and ongoing safety issues to ensure the completeness of the IPE methods and the content of PRAs for future applications.

The interdependence of the implementation program for the Severe Accident Policy with the resolutions of USIs and GSIs will not preclude the programmatic resolution of these safety issues generically.

6.2.3 PRA Reviews and Insights Reports

The implementation program relies heavily upon the lessons learned out of the NRC's experience from the reviews of past PRAs and PRA-types of analyses. Past NRC efforts were directed toward the specific risk reviews of the high-population-density plants, i.e., Indian Point, Zion, and Limerick.

Additional NRC efforts were directed specifically toward the risk portion of the review for the GESSAR II design Final Design Approval (FDA). The review of the GESSAR II design provided a special incentive to address severe accident phenomena. The staff efforts on the GESSAR II FDA and the Implementation Program will jointly consider severe accident phenomena including the concerns recently identified by the ACRS.[13]

The NRC's summary experience has been documented within the Insights Reports beginning in December 1984.[21,22] The Probabilistic Safety Analysis Procedures Guide[10] was prepared to provide the structure of a probabilistic safety study and indicates the products useful to regulatory decisions. The Probabilistic Safety Analysis Review Manual[14] was prepared to guide the NRC during the review and interpretation of risk assessments. These documents provide a formal basis to rely upon.

However, during the informal work within the implementation program each task will be undertaken jointly among several NRC branches to continue profiting from the NRC's collegial experience. In addition to the work on unresolved safety issues, inter-branch work is planned for the tasks on the IPEs, the role of PRAs in future applications, the containment performance design objectives, and the uncertainties analyses.

6.3. Industry Programs

Two groups within the nuclear industry have been particularly responsive to NRC considerations of severe-accident issues and have enhanced their

expertise on severe-accident phenomena. The implementation program calls for continued interfaces with the appropriate industry groups to supplement the expertise available to the staff on the complex of severe accident issues. Industry groups add an implementation perspective based upon more familiarity with the design and operational details of the plants. The interfaces with both industry groups are described below.

6.3.1. The Industry Degraded Core Rulemaking Program (IDCOR)

The IDCOR program has contributed in two important ways to the development of a systematic approach to Individual Plant Examinations (IPE).

First, they have developed a detailed risk evaluation method and applied it to four plants. Over the past two years, the NRC staff and contractors have met with IDCOR at regular intervals to critique the methods and results. Numerous technical issues have been resolved and several others remain to be resolved. A number of differences remain between the IDCOR severe accident analyses and those performed by the NRC-sponsored SARP PROGRAM. A process has been established to facilitate resolving the remaining issues in 1986. The IDCOR methods and risk analyses have provided several important insights into severe accident sequences, accident management and accident phenomenology. The NRC/IDCOR technical exchange will continue throughout the course of this program, and the IDCOR results will be factored into the development of guidelines and criteria.

Second, IDCOR has developed and tested a simplified methodology to be used by utilities for the individual plant examinations. The method has been applied to three BWRs and four PWRs. The NRC staff intends to review the simplified methodology with the intention of approving it, with modifications, for use in the IPEs. The methodology is scheduled for submittal, along with details on its application to seven plants, early in 1986.

6.3.2 Atomic Industrial Forum Interface

An interface is being established between the NRC staff and the Atomic Industrial Forum (AIF) to provide a forum for industry input as well as technical exchanges on source term issues. The AIF cognizant group in this effort is the Ad Hoc Special Action Group on Regulatory Applications of Source Terms, cochaired by R.P. Schmitz and S. Bernsen, which in turn is working through the AIF Committee on Reactor Licensing and Safety, chaired by M. Edelman.

A preliminary meeting was held in late October 1985 and the additional technical information meetings are planned. The next meeting is scheduled in February 1986 for the staff and AIF to exchange information concerncerning (1) the selection of a regulatory principle, (2) the removal of containment spray additives, and (3) the amount of credit to be given for decontamination by BWR suppression pools. These meetings are anticipated to continue on an as needed basis.

An additional cognizant group interface is anticiapted between the NRC staff and the AIF on PRAs. The added interface would facilitate technical exchanges during the develcpment of NRC guidance on the role of PRAs in future applications.

7. Limitations and Potential Problems

This Section delineates (1) those items associated with the severe accident issue that will not be resolved by the implementation program and (2) those contingencies that could occur during the completion of the implementation program.

The current implementation program excludes from the IPEs the assessment of the plants against external events, i.e., hazards such as earthquakes, external floods, and extreme winds. However, several sections of the Standard Review Plan for younger plants and the Safety Program Evaluation topics for older plants include acceptance criteria to assure public safety against external events. Some current PRAs that assessed the risk from external events seemed to conclude that compliance with the deterministic acceptance criteria for external events provides plants with the capacity to sustain external events somewhat more severe than the Design Basis Events.

Within the Severe Accident Policy Implementation Program, different potential approaches were considered for external events and are discussed in a draft Commission Paper scheduled to be issued early in 1986 (Subject: Treatment of External Events in the Implementation of the Severe Accident Policy Statement). The staff is inclined toward an approach that treats (1) selected seismic events, (2) all internally initiated floods and fires, and (3) selected externally initiated events. The certain seismic events are limited to relatively likely earthquakes (up to three or four times the Safe Shutdown Earthquake) and to a maximum ground acceleration of 0.5g. The limit on ground acceleration will

prevent examination of a few plants located in high seismic acceleration zones. These plants will require special attention. Furthermore, the seismic examination of individual plants should focus on finding and correcting vulnerabilities. Resolution of the question whether seismic risk is comparable to risk caused by internal initiators should not be part of the effort. Internally initiated floods and fires are included in the IDCOR study. Similarly, the NRC analysis of the reference plants should also address these events. High winds and externally initiated floods and fires should be included in the plant examination only if the specific plant site warrants it.

It must be noted that the implementation of the Severe Accident Policy depends upon the state of the art both in PRAs and in severe accident phenomenology research. The state of the art in severe accident phenomenology is evolving as the active research community continually generates new information.

Interpretations of the new information can vary and require techncial assessments. We will make every feasible effort to include the important new research information as it becomes available. The IPEs will accommodate the open issues on the phenomenlogy of degraded core conditions and the progression of a postulated severe accident. However, the IPEs by themselves will not provide the scientific resolution of these issues. The intention is to apply to regulatory practices the best available information on severe accidents which has already accumulated.

The state of the art of PRA methods and applications continues to evolve. Some diversity exists among current PRA methods and applications.[23,24] PRAs will remain subject to concerns over modeling completeness. During the implementation program, special consideration will be given to the issues among current PRAS concerning the treatment of operator errors, common mode failures, systems interactions, and the characterization of data for component failure rates and systems unavailabilities.

The implementation program depends upon precise coordination among the various organizations. Thus, the schedules within the plan are inherently vulnerable to delays. Some of the most important inputs for the schedule are on the analyses of the six reference plants (see Section 6.1).

The current approach to the treatment of sensitivities and uncertainties in the reference plant analyses is new. Because this work is partly under development the potential exists that delays in completing the sensitivity and uncertainty analyses might delay the implementation program relative to the present schedule.

There are few safety-related structures, systems and components in a nuclear power plant to which the severe accident issue is not related. The evaluation of each safety-related structure, system, and component from a severe-accident perspective could reveal previously undiscovered safety issues. If such an exigency occurs, it will be examined to determine the urgency of its impact upon public health and safety for immediate disposition in addition to its consideration within the implementation program.

8. References

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