Enclosure (1)
LD-88-005

COMBUSTION ENGINEERING, INC.

SYSTEM 80+™ STANDARD DESIGN

DESIGN CERTIFICATION

LICENSING REVIEW BASES

JANUARY 18, 1988

EXECUTIVE SUMMARY

Combustion Engineering has announced to the U.S. Nuclear Regulatory Commission its intention to pursue a Design Certification for the System 80+™ Standard Design. This effort will proceed on a new docket that will be established with all the past history and safety evaluation (including the FDA) of the current System 80, as described in CESSAR-F, as the starting point. The design enhancements and expanded scope for the System 80+ Standard Design will be fully described in CESSAR-Design Certification (CESSAR-DC) and are intended to yield a standard plant design that not only meets all current regulations but also satisfies the criteria of the Commission's Severe Accident and Standardization Policy Statements.

In the absence of fully defined acceptance criteria for the review of standard plant designs against the Severe Accident and Standardization Policy Statements, these Licensing Review Bases will serve to (1) outline the development of appropriate acceptance criteria for key areas of the Staff's review of the System 80+ Standard Design and (2) establish a clear definition of the schedule, process and administrative matters which will be used to review and certify the System 80+ Standard Design.

The System 80+ Standard Design includes the Nuclear Power Module plus Standardized Functional Descriptions. The Combustion Engineering scope of supply is the Nuclear Power Module and is a major portion of a complete nuclear power plant design. Combustion Engineering has also committed, however, to the provision of a sufficient level of detail on the remaining portions of the plant design via detailed Standardized Functional Descriptions to allow the Staff to make a complete and conclusive public health and safety determination for the System 80+ Standard Design. The Staff's review of CESSAR-DC, therefore, will close out all questions concerning the System 80+ Nuclear Power Module and will fully establish the requirements for the remaining portions of the Standard Design.

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1.0 INTRODUCTION

Combustion Engineering has announced its intention to pursue a Design Certification in accordance with the Commission's Nuclear Power Plant Standardization Policy Statement of September 15, 1987.

The Commission's Standardization Policy Statement (52FR34884) declares that future reference system designs "are expected to be evolutions of existing proven LWR designs". Accordingly, Combustion Engineering is enhancing the System 80^{R} standard design to meet the requirements of the NRC's Severe Accident and Standardization Policy Statements. The scope of the improved design, called the System 80+^{TM} Standard Design, will include the Nuclear Steam Supply System, the emergency feedwater system, the containment, and the control room (collectively refered to as the Nuclear Power Module) as well as detailed Standardized Functional Descriptions for all other systems requiring regulatory review. This expanded scope, depicted in Figure 1, will provide sufficient information to enable the Staff to conclusively reach the required public health and safety determination for the System 80+ Standard Design.

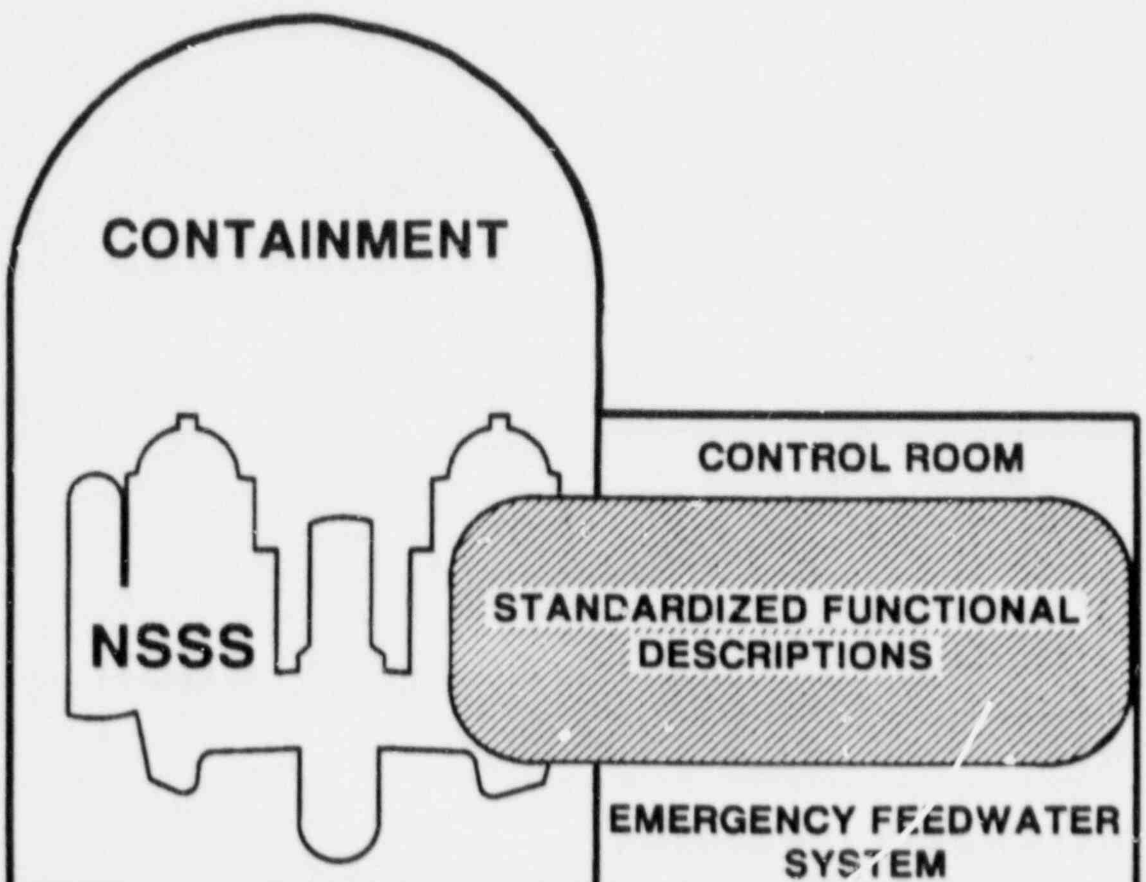
FIGURE 1
SCOPE OF THE SYSTEM 80+ STANDARD DESIGN

NUCLEAR POWER MODULE

1. Reactor Coolant System
2. Safety Injection System
3. Containment Isolation System
4. Engineered Safety Features Actuation System
5. Fuel Handling System
6. Chemical and Volume Control System
7. Shutdown Cooling System
8. Containment Spray System
9. Reactor Protection System
10. Control Systems
11. Monitoring Systems
12. Nuclear Instrumentation
13. Control Room
14. Containment Building
15. Emergency Feedwater System
16. Safety Depressurization System
17. Main Steam and Feedwater Instrumentation and Component Controls

STANDARDIZED FUNCTIONAL DESCRIPTIONS

Detailed descriptions for all other plant systems to enable the Staff to reach the required public health and safety determination for the System 80+ Standard Design.



The NRC Staff believes that the safety review of CESSAR-DC will proceed more smoothly if certain licensing review bases are established as early as possible. This Licensing Review Bases (LRB) document will, therefore, be used to outline the development of acceptance criteria for key areas of the Staff's review of System 80+ and to establish a clear definition of the schedule, process and administrative matters which will be used to review and certify the System 80+ Standard Design. The LRB, in conjunction with the acceptance criteria to be developed, is intended to serve as guidance for the NRC Staff review of material submitted in compliance with criteria that go beyond current regulations (e.g., the Severe Accident Policy).

The development of LRB is particularly important because:

(1) a Design Certification process has not yet been fully defined by the Commission,
(2) System 80+ will be the first PWR standard design to proceed to Design Certification,
(3) the System 80+ Standard Design will include features not required by the existing rules and regulations of the Commission as defined by the Severe Accident Policy Statement,

(4) review procedures and acceptance criteria have not yet been established for a standard plant PRA, and

(5) acceptance criteria have not been fully established for the resolution of Unreviewed Safety Issues/Generic Issues (USIs/GIs) and degraded core issues.

The staff fully supports the efforts of Combustion Engineering to obtain Design Certification for the System 80+ Standard Design. Once the design has been certified, it can be referenced by a number of applicants for use on a number of different sites without further design review.

To accomplish this objective, the design must be described in sufficient detail to ensure that all regulatory matters at issue are adequately addressed and closed prior to completion of the Design Certification process. This would ensure that, when an applicant references the certified design, the staff can limit its review to a compliance review which would confirm that the plant was built in accordance with the System 80+ Standard Design (the Nuclear Power Module and the Standardized Functional Descriptions) established and certified in CESSAR-DC.

1.1 Scope and Content of CESSAR-DC

The System 80+ Standard Design will use, as a starting point, the design covered by the current FDA and described in CESSAR-F. By utilizing this "FDA Design", Combustion Engineering is starting with a reference design which already complies with current NRC regulations and requirements for existing plants. This compliance is highlighted by the fact that Palo Verde Units 1, 2 and 3 referenced the CESSAR FDA in their successful operating license applications.

The expansion of the System 80 design to include the Nuclear Power Module and detailed Standardized Functional Descriptions will ensure that adequate information is provided to the Staff to enable all safety issues for the System 80+ Standard Design to be fully addressed and closed during the Design Certification Process. Furthermore, experience in the previous review of System 80 for its current Final Design Approval (FDA) provides reasonable assurance that the Staff can receive all of the information needed to complete its review of the System 80+ Standard Design with a level of detail sufficient to close out all applicable regulatory review issues.

Since the objective of this program is to certify the System 80+ Standard Design prior to identification of the utility applicant, the site or sub-suppliers, it is necessary that the level of detail [illegible]e sufficient information to enable the Staff to complete its review without posing anti-competitive constraints. Prior CESSAR-F experience has shown that this should not represent a limitation on the Staff's ability to complete its review. The depth of design information needed to conduct this review is the level which demonstrates compliance with NRC regulations sufficient to close out all applicable safety issues.

As required by the Severe Accident Policy Statement (50FR32138), CESSAR-DC will describe System 80+ changes required to demonstrate the technical resolution of all applicable Unresolved Safety Issues, the medium- and high-priority Generic Issues, and other issues identified in the Severe Accident Policy Statement. As discussed in the previous paragraph, CESSAR-DC will contain sufficient information to permit the Staff to complete its review of the System 80+ Standard Design and, hence, to resolve all applicable safety issues.

1.2 Scope and Content of Future Applications Referencing CESSAR-DC

When the certified System 80+ Standard Design is referenced in an application, the Staff's review of matters related to the approved design need consider only whether the site envelope parameters and the Standardized Functional Descriptions have been satisfied in the referencing application. Specifically, for the site envelope and those areas in the remainder of the plant where CESSAR-DC has specified Standardized Functional Descriptions, the applicant will only have to demonstrate compliance with them. No further review of the referenced design itself (the System 80+ Standard Design) will be required when the site envelope parameters fall within the design envelope and all Standardized Functional Descriptions are satisfied.

2.0 SCHEDULE

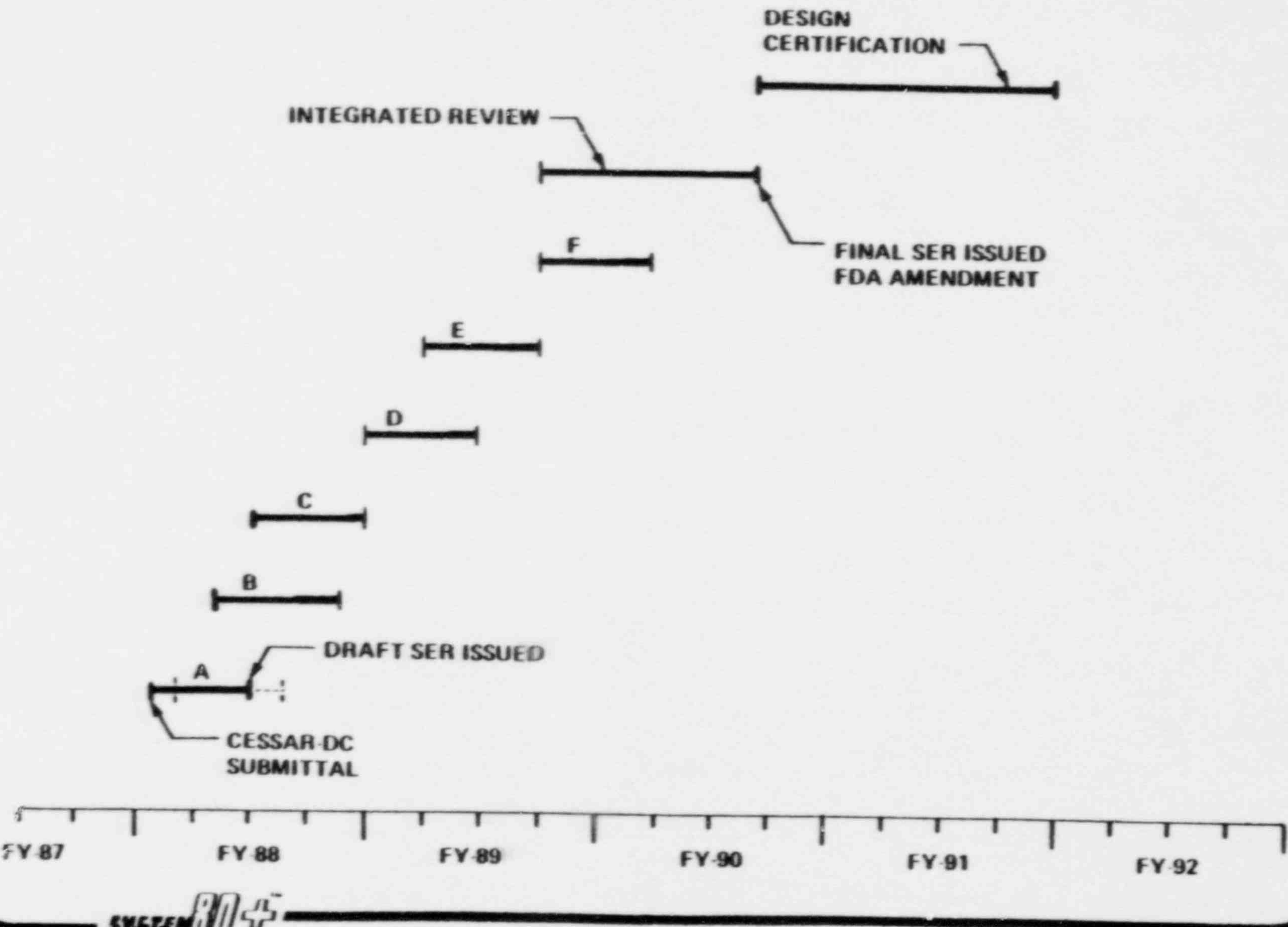
The schedule for submitting groups of CESSAR-DC chapters is shown in Table 1 and the schedule for NRC review of those submittals is shown in Figure 2. The review schedule shows an average review period of six months for each submittal group. This is an appropriate review period for CESSAR-DC chapters which describe the NSSS since the NSSS is based on System 80 which has already been reviewed and approved. Additional time may be required for review of the expanded scope items (the control room, emergency feedwater system and the containment). To facilitate meeting this schedule, early meetings will be encouraged and any resulting schedule commitments will be documented by NRC in "meeting minutes" memoranda.

Table 1
CESSAR-DC Submittal Schedule

Submittal Group	Description	Revision of CESSAR-DC Chapter (Sections)	Implementation of EPRI Chapter	CESSAR-DC Submittal Date	Draft SER Issues
A1	General Descriptions and Requirements	1 (all)	1	Submitted Sept. 1987	March 1988
A2	Power Conversion System, Quality Assurance	10 (10.1,10.3,10.4) 17 (all)	2	Submitted Nov. 1987	May 1988
B	Reactor Coolant System, Chemical and Volume Control System, Process Sampling System, Boron Recycle System	4 (all) 5 (5.1,5.2,5.4) 9 (9.3)	3 & 4	Feb. 1988	August 1988
C	Reactor Coolant System, Emergency Feedwater System, Safety Injection System, Shutdown Cooling System	3 (3.1,3.2,3.6,3.9) 5 (5.3,5.4,5A,5B,5C) 6 (6.1,6.3,6.6,6.7) 10 (10.4)	5	March 1988	Sept. 1988
D	Building Design & Site Arrangements, Instrumentation & Control Systems, Human Factors Engineering	2 (all) 3 (3.3-3.5,3.7,3.8,3.10 3.11,3A) 5 (5.4) 6 (6.2,6.4,6.5,6A,6B) 7 (all), 18 (all)	6 & 10	Sept. 1988	March 1989
E	Fuel Handling Systems, Radioactive Waste Systems	8 (all) 9 (9.1,9.2,9.4,9.5) 10 (10.2,10.4) 11-14 (all)	7 - 13 (except 10)	Dec. 1988	June 1989
F	Safety Analyses, Probabilistic Risk Assessment, Technical Specifications	6 (6.3) 15-16 (all) Appendices (all)	-	June 1989	Dec. 1989
-	Integrated Review	All	-	June 1989	June 1990
-	Receive FDA Amendment	-	-	-	June 1990
-	Receive Design Certification	-	-	-	Sept. 1991

FIGURE 2

NRC REVIEW SCHEDULE



3.0 CONTENT OF APPLICATION

3.1 Dual Docket Approach

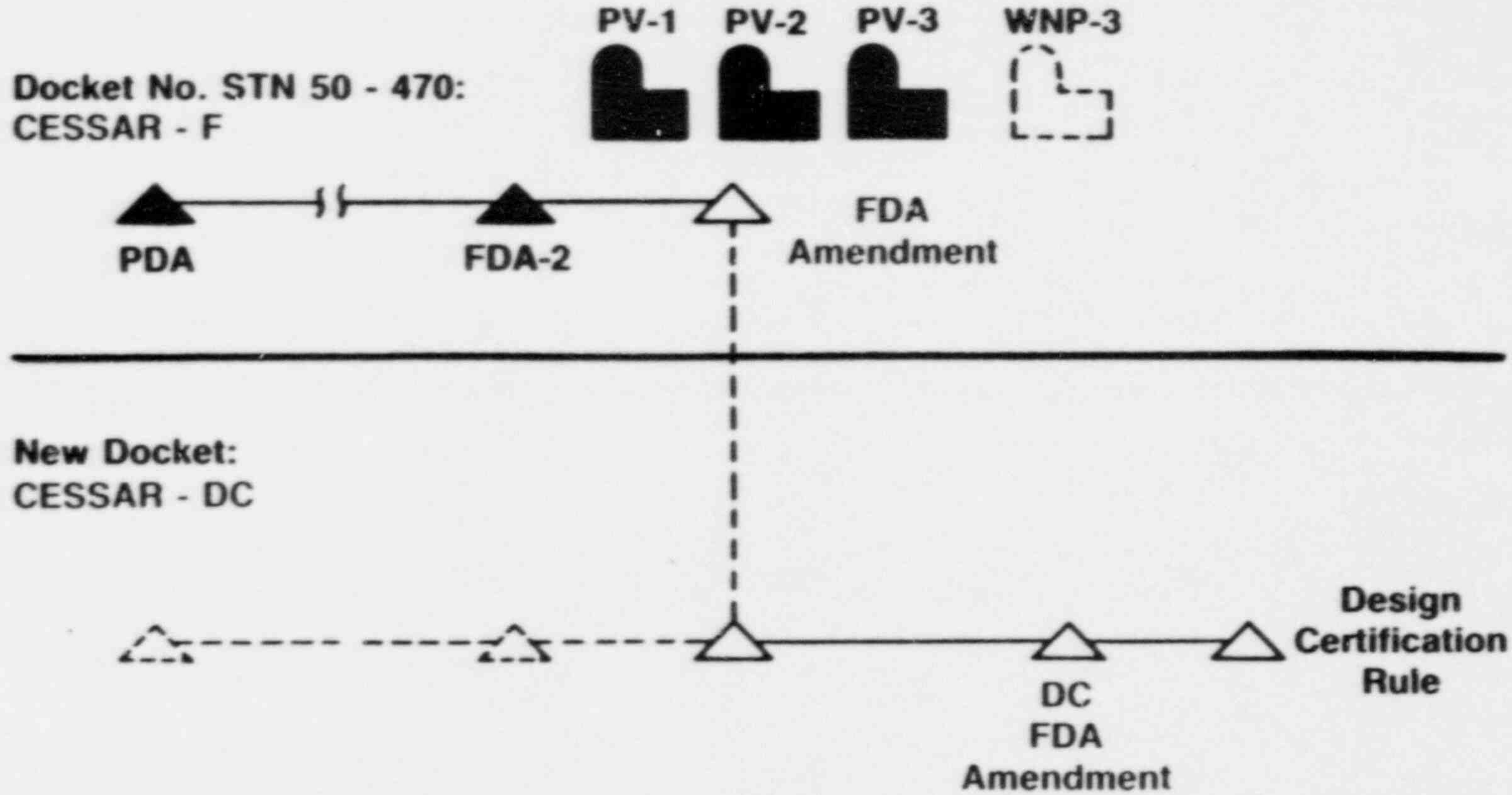
A second (separate) docket will be created which includes all of the existing information and history of the current System 80 docket, docket number STN 50-470. As shown in Figure 3, the new docket will be utilized to describe the System 80+ Standard Design and to, thus, provide the basis for the Design Certification Rule. This approach will allow current System 80 users to reference the first (current) docket while, at the same time, allowing for development of the System 80+ Design Certification Rule.

3.2 CESSAR-DC Format

The safety review of the System 80+ Standard Design for Design Certification will be performed by NRC reviewers who are accustomed to working with the format and organization of the NRC's Standard Review Plan (NUREG 0800) and Regulatory Guide 1.70 (Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants). CESSAR-F has already been reviewed and approved by the NRC Staff for the current FDA. Combustion Engineering will, therefore, make revisions in CESSAR-DC in a format consistent with past review experience and in full compliance with Section B.3.b of the Commission's Severe Accident Policy Statement.

FIGURE 3

DUAL DOCKET APPROACH



3.3 CESSAR-DC Amendment Identification

The CESSAR-DC submittals outlined in Table 1 will consist of changes to existing CESSAR-F material in Chapter-by-Chapter packages. Bars with amendment identifiers will be provided in the margins to indicate all areas of change relative to CESSAR-F and the CESSAR-DC amendment identifier and date will be provided at the bottom of each amended page.

4.0 INCORPORATION OF NEW ISSUES

As stated in the Severe Accident Policy Statement (Section 7), the Commission expects that future plant designs will meet current regulations and will address new issues such as the resolution of USIs and GIs, Probabilistic Risk Assessment, and degraded core analyses. Combustion Engineering will address these new issues such that there are no open items when the NRC issues the FDA Amendment for CESSAR-DC (see Section 7 and Appendices A, B, and C for more detail). Combustion Engineering is committed to full implementation of the Severe Accident Policy Statement and will include resolutions for all applicable USIs and High- and Medium-priority GIs in the System 80+ Standard Design.

By issuing FDA-2 to Combustion Engineering and by issuing Operating Licenses to Palo Verde Units 1, 2, and 3, the Commission recorded its determination that the System 80 design meets the existing rules and regulations of the Commission and provides adequate protection to the health and safety of the public. Since the System 80 design has already received an FDA, the requirements of the Backfit Rule apply to NRC-required revisions to the design beyond those sponsored by Combustion Engineering. That is, the final regulatory standard for Staff required changes beyond those offered by Combustion Engineering will be the cost-benefit provisions of the Backfit Rule. Combustion Engineering will be required to make additional changes to the design only if analyses show that the costs of such changes are justified by the increase in the overall protection of public health and safety that would be provided.

5.0 REVIEW PROCEDURES

5.1 Overview of NRC Staff Review

Each NRC reviewer will be provided a complete copy of the CESSAR-F Safety Evaluation Report (NUREG 0852 and supplements). After reviewing this report, NRC Staff will review the design changes identified in CESSAR-DC to confirm that compliance with NRC rules and regulations remains valid; that is, equal to or more conservative than what is stated in the existing CESSAR-F Safety Evaluation Reports. The NRC Staff will then confirm that the design changes comply with the guidance of the Standardization and Severe Accident Policy Statements.

Proposed acceptance criteria and design features suitable for resolution of all applicable USIs and High- and Medium-Priority GIs will be proposed and documented by Combustion Engineering in an appendix to CESSAR-DC. The NRC Staff will review the acceptance criteria and proposed resolutions to these USIs and GIs on a schedule consistent with NRC review of CESSAR-DC chapters (Section 2).

Combustion Engineering has committed to the provision of a sufficient level of information [through detailed Standardized Functional Descriptions (Section 8.3)] to allow the NRC Staff to complete its review of the System 80+ Nuclear Power Module and conclusively reach the required public health and safety determination. NRC Staff acceptance of the Standardized Functional Descriptions will fix the requirements for the remaining portions of the Standard Design outside the scope of the Nuclear Power Module.

5.2 Format of Safety Evaluation Report

Because CESSAR-DC will be submitted as revisions to groups of chapters over a two year period, the Safety Evaluation Report will be issued initially in draft form and in sections (see Section 2, Figure 2 for the schedule). It will be important to carefully document open issues that may be identified in the review process which cannot be resolved until the completion of later chapters. Each draft SER section will contain a full description of such issues and will be issued at the completion of the Staff's review of each submittal. The Staff's final Safety Evaluation Report for the System 80+ Standard Design will be issued after an integrated review is completed and will be in the same form used for other reactor licensing applications. [Combustion Engineering will maintain an updated checklist which identifies outstanding issues and the future chapter(s) in which resolution is anticipated. Open items identified by the Staff will be added to this tracking list by Combustion Engineering].

With respect to USIs and GIs, the draft SERs will address the acceptance criteria and proposed resolution and provide the Staff's preliminary concurrence as appropriate. Staff approval of these criteria and resolutions will be finalized when all CESSAR-DC chapters have been submitted and the integrated review has been completed.

5.3 Questions and Responses

As the Staff's review progresses, there is likely to be a need for additional information from Combustion Engineering. The NRC procedure to be used is described below. This procedure will be applied to the resolution of all NRC questions. To improve the efficiency of the review, the NRC Staff encourages informal communication while assuring that resolution of issues is formally documented. Throughout this process all written (formal) communications to Combustion Engineering will be directed to the Director of Nuclear Licensing and all informal communications will be directed to the Manager of Standard Plant Licensing. The steps are as follows:

1. After a CESSAR-DC submittal is received, reviewers will be expected to review the revisions in detail and, if necessary, submit requests for additional information (RAIs) to the NRC Project Manager (PM). Key RAI items will be submitted within one month and a complete RAI within two months after the CESSAR-DC chapters have been received.

2. The NRC PM will compile the RAIs as they are received, and transmit them immediately to Combustion Engineering. Through informal communications with Combustion Engineering, such as conference calls, the RAIs will be further reviewed with some being answered informally and/or withdrawn. Some RAIs may include requests for information that is not expected to be available until the submittal of a later CESSAR-DC chapter. These RAIs will be deferred to future chapter submittals and the draft SER will be written accordingly. A final RAI transmittal to Combustion Engineering will be completed within two weeks after the RAIs have been submitted to the NRC PM (about two months after submittal of CESSAR-DC chapters).

3. Combustion Engineering and the NRC Staff will mutually agree on a meeting schedule. The meetings are expected to begin during the third month after submittal of each CESSAR-DC chapter and should be completed during the fourth month.

4. The NRC PM and the reviewer(s) will document the results of the meetings and Combustion Engineering will formally respond to the final RAI by the end of the fifth month.

5. Staff reviewers will be expected to complete their sections of the SER within one more month so that a draft SER for each CESSAR-DC submittal will be available within six months.

Every effort will be made to make the first round of RAIs the only one necessary. If the NRC Staff believes a second round of RAIs is necessary, however, the same procedure will be followed, but it is expected that a shorter schedule will be used. For the first round of RAIs, the above schedule shows a 6 month review for each CESSAR-DC submittal group listed in Table 1. If a second round of RAIs is necessary, however, a total of approximately nine months may be required.

5.4 Integrated Review

At the completion of the review of the individual CESSAR-DC chapters, the same staff reviewers who conducted the individual chapter reviews will perform an integrated review of the complete CESSAR-DC to ensure all open review issues are resolved. This review will complement the PRA and safety analysis reviews, in that it will be an overall assessment of the design. The Staff will issue a composite final SER in accordance with the schedule described in Section 2. There will be no open issues at the completion of the NRC Staff review and issuance of the FDA amendment.

6.0 ACRS PARTICIPATION

One step in the design review of a standard pl   is the independent review by the Advisory Committee on Reactor Safeguards (ACRS). The NRC PM will keep the ACRS informed of the progress of the review and will forward copies of CESSAR-DC chapters as they are submitted, along with copies of the draft SERs as they are issued. In addition, the NRC PM will schedule a meeting with the ACRS to discuss the final SER.

7.0 SEVERE ACCIDENT POLICY

7.1 Introduction

On August 8, 1985, the Commission issued a policy statement on severe accidents (50FR32138, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants"). The policy statement provides general criteria and procedures for the licensing of new plants, and sets goals and a schedule for the systematic examination of existing plants. The Commission encouraged the development of new designs that might realize safety improvements and stated that the Commission intended 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. The Commission's general licensing criteria for future plants are specified in the policy statement.

The Commission further recognized the need to strike a balance between accident prevention and consequence mitigation, through a better understanding of containment performance, with the understanding that new performance criteria for containment systems might need to be established. The Commission also recognized the importance of potential contributors to severe accident risk such as human performance and sabotage, and determined that these issues should be carefully analyzed and considered in the design and operating procedures for the facility. As indicated below, Combustion Engineering will meet the guidance specified for new plants.

7.2 Compliance With General Licensing Criteria

7.2.1 TMI Requirements for New Plants

Combustion Engineering will comply with all regulations applicable to the System 80+ Standard Design which are listed in 10 CFR 50.34(f).

7.2.2 Resolution of USIs and GSIs

The process for developing the resolution of USI's and GI's is provided in Appendix A.

7.2.3 Probabilistic Risk Assessment

The process of preparing and using the System 80+ Standard Design PRA is provided in Appendix B.

7.2.4 NRC Staff Review

The approach to the Staff review of CESSAR-DC is described in Sections 2 through 5 of this document. The process for the review of degraded core analyses complemented by PRA is discussed in Appendix C.

7.3 Severe Accident Performance Goals

This section describes the goals, or approximate values, for severe accident performance criteria. These goals are consistent with the guidance of the NRC's Safety Goal Policy. The NRC Staff will use these goals during the review of the System 80+ Standard Design, but they will be considered as guidance, not as firm criteria.

One of Combustion Engineering's objectives for the development of the System 80+ Standard Design is to be responsive to utility requirements for increased public safety and protection of plant investment. The approximate goals stated in the following sub-sections were developed to meet those utility requirements while remaining consistent with NRC guidance.

7.3.1 Prevention of Core Damage

Compliance with current regulations provides adequate protection of the public and the safety analyses ensure that the reactor core is protected consistent with those regulations. The EPRI ALWR Requirements Document provides Utility requirements for an improved nuclear plant. One of the broad objectives of these requirements is to provide adequate protection of plant investment. One of EPRI's criteria for increased protection of plant investment is the estimated mean annual core damage frequency target (including both internal and external events) of less than 1 X 10^{-5} events per reactor year. Another of EPRI's criteria for increased protection of plant investment is that no core damage should be predicted to occur for a near instantaneous pipe break with an equivalent diameter of six inches (using best estimate methodology).

The above EPRI criteria are being applied by Combustion Engineering as goals in the development of the System 80+ Standard Design. The actual values finally used will depend on the methodology applied and the design improvements implemented (which will be discussed with the NRC during the review of CESSAR-DC).

7.3.2 Mitigation of Core Damage

Consistent with the defense-in-depth principle, the design of the System 80+ Nuclear Power Module will provide protection against containment failure in the event of a severe accident.

The expected containment design features will include:

a. a large dry containment,

b. measures to reduce the probability of early containment failure,

c. a conservative design basis accident (guillotine pipe break),

d. severe accident hydrogen control (considering 75% active fuel-clad metal water reaction and a maximum hydrogen concentration of 13% by volume),

e. measures to prevent containment damaging hydrogen detonation,

f. an in-containment refueling water storage tank,

g. reliable containment heat removal systems, and

h. consideration of severe accidents in design of the containment and the reactor vessel cavity configuration.

7.3.3 Offsite Consequences for Severe Accidents

Compliance with current regulations provides adequate protection of the public and the safety analyses ensure that the reactor is protected consistent with those regulations. The EPRI ALWR Requirements Document provides additional Utility desires for an improved nuclear plant. Another broad objective of these requirements is to increase public safety. Accordingly, the guidance for offsite consequences will be:

> In the event of a severe accident, the dose beyond a one-half mile radius from the reactor is not expected to exceed 25 Rem to the whole body. The expected mean frequency of occurrence for higher off-site doses is expected to be less than once per million reactor years, considering both internal and external events.

The above EPRI input will be applied by Combustion Engineering as guidance in the development of the System 80+ Standard Design and not as firm criteria. The actual values finally used will depend on the methodology applied and the design improvements implemented (which will be discussed with the NRC during the review of CESSAR-DC).

8.0 ADDITIONAL ISSUES

8.1 Physical Security and Sabotage

The System 80+ Standard Design is being developed in accordance with all current NRC regulations and guidance regarding the physical security of nuclear power plants and the prevention of sabotage. It is intended that the final design be sufficiently complete in this respect, through either detailed design requirements, Standardized Functional Descriptions (Section 8.5), or general guidance supplemented by PRA results, to allow the development of a comprehensive security plan that will ensure the safety of the as-built facility will continue to be accurately described by the certified design.

8.2 Site Envelope Parameters

The System 80+ Standard Design is based on assumed site-related parameters, to be discussed in CESSAR-DC, that were selected so as to be applicable to the majority of potential nuclear power plant sites in the United States. Therefore, despite variations in site parameters from the assumed values at most specific locations, the System 80+ Standard Design can be expected to meet the necessary regulatory requirements. A nearly identical site envelope was reviewed by the NRC for the CESSAR-F FDA and the NRC concluded in the Safety Evaluation Report that the site-related information provided in CESSAR-F adequately described the site parameters postulated for the design, and that the design had been adequately analyzed and evaluated in terms of such parameters.

8.3 Completeness of Design Documentation

The level of detail of information provided in CESSAR-DC will be that which is necessary and sufficient for assuring conformance to NRC regulations and for closing out all applicable CESSAR-DC review issues.

Design documentation for systems, structures and components within the System 80+ Standard Design will include, as appropriate:

1. Design Basis Criteria
2. Plant General Arrangements of Structures and Components
3. Process and Instrumentation Diagrams
4. Control Logic Diagrams
5. System Functional Descriptions
6. Supporting Design Data
7. Quality Assurance Program
8. Design Related Aspects for the Emergency Plans

9. Design Related Aspects for the Physical Security Program
10. ALARA/Radiation Protection Plan
11. Accident Analyses
12. Technical Specifications
13. Probabilistic Risk Assessment

In a limited number of cases where detailed design information is not available, information on methods, procedures, and acceptance criteria will be provided.

8.4 Program for the Assurance of Quality in Design

The Combustion Engineering Quality Assurance Program is described in topical report CENPD-210A, Revision 4, "Quality Assurance Program", dated January, 1987, and letter LD-87-070, A. E. Scherer (C-E) to J. W. Roe (NRC), dated December 15, 1987. This program has been found to be compliant with the provisions of Appendix B to 10 CFR 50.

8.5 Standardized Functional Descriptions

In order to ensure that all applicable regulatory issues are closed out during the NRC Review and Design Certification process, the Interface Requirements (IRs) of the current CESSAR-F will be replaced by detailed Standardized Functional Descriptions (SFDs). These SFDs will provide significantly more information than the IRs of CESSAR-F.

The level of detail in the SFDs will be sufficient to enable the Staff to make the required public health and safety determinations for the System 80+ Standard Design. The SFDs will be located in CESSAR-DC consistent with the format guidance of Regulatory Guide 1.70 for "balance-of-plant" systems.

The SFDs will provide detailed descriptions for systems outside the scope of the Nuclear Power Module, which are relied upon to make safety determinations for the System 80+ Standard Design. The SFDs will identify the acceptance criteria that will ensure these safety determinations remain valid. The SFDs will begin with a discussion of the safety-related design bases of the system to which it applies. To support these design bases, the SFD will elaborate further and provide a description of the system configuration and a detailed functional description of the system features necessary to meet NRC requirements. These functional descriptions will include specific performance criteria, applicable codes and standards governing the system design, system arrangement criteria, pipe and valve performance criteria, I&C requirements, appropriate safety-related EPRI ALWR requirements, and installation requirements necessary to make the required health and safety determination for the System 80+ Standard Design. A safety evaluation will enumerate those acceptance criteria that will ensure that a system malfunction will not adversely impact Nuclear Power Module safety. Feedback from the System 80+ PRA may provide system reliability acceptance criteria which will be included in the SFDs. Additionally, the SFDs will provide the material selection requirements, fabrication requirements, testing and inspection requirements, and the appropriate chemistry requirements needed to ensure safe and reliable operation.

The goal of the SFD is to provide the Staff reviewers with a sufficient level of detail such that (1) the Staff can conduct a review of the System 80+ Standard Design and close out all applicable regulatory review issues and (2) the Staff's review of a future application referencing the System 80+ Standard Design can be limited to a simple compliance review (i.e., a review to confirm that all systems meet the certified acceptance criteria and interface requirements enumerated in the SFDs).

8.6 Instrumentation and Controls

The standards and criteria to be used by Combustion Engineering in the design of Instrumentation and Control Systems and by the Staff in the review of these systems are presented in Appendix D.

8.7 Generic Letters and IE Bulletins

Combustion Engineering will evaluate lists of Generic Letters and Inspection and Enforcement Bulletins (IE) for possible consideration in the System 80+ Standard Design. This will help ensure that all potential Staff concerns are addressed in the design process.

8.8 Maintenance and Surveillance

The development of detailed design requirements, standard technical specifications and Standardized Functional Descriptions, supplemented by an evaluation of PRA results, will ensure that sufficient maintenance guidance will be made available to the utility applicant. This documentation will allow the development of a comprehensive maintenance program that will ensure that the safety of the as-built facility will continue to be accurately described by the certified design.

8.9 Safety Goal Policy Statement

On August 4, 1986, the Commission published a Policy Statement on "Safety Goals for the Operation of Nuclear Power Plants" (51 FR 28044). This policy statement focuses on the risks to the public from nuclear power plant operations. Its objective is to establish goals that broadly define an acceptable level of radiological risk.

The implementation guidance that is developed by the Staff will -- as appropriate -- be applicable to the System 80+ Standard Design. Combustion Engineering will apply the severe accident performance goals of Section 7.3 as approximate criteria (or targets) during the design and analysis of the System 80+ Standard Design.

8.10 Standardization Policy Statement

Consistent with the Commission's Standardization Policy Statement, Combustion Engineering's System 80+ Design Certification Program emphasizes the development of a standard design based on the evolution of a proven technology. The System 80+ Design Certification Program will be conducted in accordance with the Standardization Policy and any final Standardization Rule established by the Commission. It will be necessary, however, for the NRC Staff to keep Combustion Engineering informed concerning the nature of the pending Standardization Rule to avoid last minute delays.

9.0 FINAL DESIGN APPROVAL

In August 1985, Combustion Engineering requested that the CESSAR FDA (FDA-2) be amended to permit forward referenceability in accordance with the NRC Severe Accident Policy Statement. Upon completion of NRC Staff review of that request, the Staff will issue a forward referenceable FDA Amendment that will be applicable to both dockets, as described in Section 3.1.

When the NRC Staff completes its review of CESSAR-DC, the FDA (on the new docket only) will be amended again to document the closeout of all applicable NRC review issues for the System 80+ Standard Design. The amended FDA will be the basis for a System 80+ Design Certification Rule.

10.0 DESIGN CERTIFICATION

As indicated in the Standardization Policy Statement, the Commission believes that the use of pre-approved standard plant designs can benefit public health and safety by:

1. Concentrating the resources of designers, engineers, and vendors on particular approaches;

2. Stimulating standardized programs of construction practice and quality assurance;

3. Improving the training of personnel; and

4. Fostering more effective maintenance and improved operation.

The use of such pre-approved standardized designs can also permit more effective and efficient licensing and inspection by the NRC. The Design Certification concept provides for certifying a reference system design, such as the System 80+ Standard Design, through rulemaking. In this process, the Commission would certify a design after rulemaking proceedings are completed. The Design Certification means that the System 80+ Standard Design has been found acceptable for incorporation by reference in an individual license application. The conclusions of the certification rulemaking would be used and relied upon by the NRC Staff, the ACRS, the hearing boards, and the Commission in their reviews of applications that reference the design.

Combustion Engineering's Design Certification Program will be conducted in accordance with the Commission's Standardization Policy and any final Standardization Rule established by the Commission.

APPENDIX A

Combustion Engineering Design Certification Program

Process for Resolution of Unresolved and
Generic Safety Issues as Required by the
Severe Accident Policy Statement.

I. Overview of Process for Resolution of USIs and GIs

One of the major goals of Combustion Engineering's Design Certification Program is to develop and obtain NRC certification of a standard design (the System 80+™ Standard Design) which meets the requirements of the Severe Accident Policy Statement (SAPS) for future plants. In order to comply with the SAPS, technical resolution of all applicable Unresolved Safety Issues (USIs) and Medium- and High-Priority Generic Issues (GIs) must be demonstrated for the System 80+ Standard Design.

Combustion Engineering will integrate input from related industry programs (e.g., the EPRI Regulatory Stabilization Program) and implement resolutions to the USIs and GIs for the System 80+ Standard Design. Documentation of the acceptance criteria and design features for resolution of the USIs and GIs will be provided in an appendix to Combustion Engineering's Standard Safety Analysis Report - Design Certification (CESSAR-DC). It is expected that NRC Staff will request from Combustion Engineering information necessary to close out all applicable review issues so that a Design Certification rulemaking can be concluded without any open issues or conditions.

II. Acceptance Criteria for Resolution of USIs and GIs

The USIs and GIs that are required to be addressed for compliance with the SAPS are identified in the NRC's Generic Issue Management Control

System (GIMCS). Some of the issues in GIMCS await prioritization (59 as of June 1987). Others have been prioritized into categories of USI, and High-, Medium-, and Nearly Resolved Generic Issues. Based on the GIMCS listings, the C-E Design Certification Program will identify and resolve the USI's and the High- and Medium-Priority GI's which are found to be applicable to the System 80+ Standard Design. A preliminary list of applicable issues is presented as an attachment to this appendix.

In order to resolve the applicable USIs and GIs, proposed acceptance criteria must first be documented (by either the NRC or by an applicant). Then, resolutions must be proposed and reviewed by NRC Staff. Combustion Engineering will integrate input from various sources (described below) and will coordinate all activities required to prepare proposed acceptance criteria and the corresponding resolutions. Each applicable issue will be resolved and documentation will be submitted on the CESSAR-DC docket. Some issues have already been resolved by the NRC and Combustion Engineering will implement, to the maximum extent possible, the NRC's documented resolutions. If, however, some revisions are necessary, Combustion Engineering will propose alternate resolutions appropriate for the System 80+ Standard Design.

Some issues have not yet been resolved. For those issues which are applicable to System 80+, C-E will review results of the EPRI Regulatory Stabilization Program and DOE's Advanced Reactor Severe Accident Program (ARSAP). To the maximum extent practical, results from these programs will be implemented for the System 80+ Standard Design.

The EPRI Regulatory Stabilization Program is developing Topic Papers on proposed acceptance criteria for resolution of the USIs and GIs which are applicable to Advanced LWR designs. The primary purpose of these Topic Papers is to document criteria for resolution of applicable issues and incorporate NRC review comments into those criteria. The C-E Design Certification Program will address and resolve the USIs and GIs via design features which are expected to be consistent with the criteria in the Topic Papers. In this way, the issues can be closed out by NRC, based on documented criteria which have been reviewed by NRC.

Topic Papers will also be generated in the ARSAP to address severe accident issues. ARSAP staff have reviewed current information related to severe accidents to identify a composite list of related issues for which Topic Papers will be produced. Some of these Topic Papers may also be applicable to resolution of the USIs and GIs which must be resolved for the System 80+ Standard Design. For these particular USIs and GIs, C-E will integrate input from the DOE ARSAP and present the proposed acceptance criteria and resolutions to the NRC for review and comments.

There may be some USIs and GIs, however, for which Topic Papers are not available from either the EPRI Regulatory Stabilization Program, the DOE ARSAP or from the NRC. For these USIs and GIs, C-E will develop acceptance criteria and resolutions specific to the System 80+ Standard Design and will obtain NRC approval through documentation in CESSAR-DC.

III. <u>NRC Review Process and Documentation</u>

Proposed acceptance criteria and design features for resolution of applicable USIs and GIs will be documented by Combustion Engineering in an appendix to CESSAR-DC. The NRC will review this appendix and Combustion Engineering will provide any additional information necessary for preliminary NRC concurrence. Final NRC approval of the proposed resolutions will occur as part of the Design Certification rulemaking. Combustion Engineering will provide sufficient information in CESSAR-DC so that the appendix can serve as the primary documentation of acceptance criteria for USIs and GIs during NRC Staff and ACRS reviews.

The NRC will review the acceptance criteria and proposed resolutions to specific USIs and GIs on a schedule consistent with NRC review of the chapters of CESSAR-DC. The schedule for CESSAR-DC submittals to the NRC is provided in Section 2 of this Licensing Review Basis document.

NRC review results will be documented in draft Safety Evaluation Reports (SERs) on the schedule described in Section 2 of this document. The draft SERs will address the acceptance criteria for the USIs and GIs, as well as the resolutions (design features) proposed for the System 80+ Standard Design. NRC's preliminary concurrence with the acceptance criteria and resolutions will be provided in the draft SERs. The draft SERs will be finalized when all CESSAR-DC chapters have been submitted and an integrated review has been completed by the NRC Staff.

IV. Summary

Combustion Engineering's Design Certification Program for the System 80+ Standard Design will resolve all applicable USIs and GIs, as required in the Severe Accident Policy Statement. Input from related industry programs and existing NRC documentation will be reviewed and integrated in order to identify acceptance criteria for resolution of the USIs and GIs.

The resolution of USIs and GIs for System 80+ will be based primarily on acceptance criteria from EPRI ALWR and DOE ARSAP Topic Papers. C-E will integrate these inputs and develop additional criteria, if and where necessary. Documentation of the acceptance criteria and proposed design features for resolution of all applicable USIs and GIs will be provided in an appendix to CESSAR-DC. Combustion Engineering will provide whatever information is necessary to close the USIs and GIs for the System 80+ Standard Design. NRC's preliminary concurrence with the acceptance criteria and proposed resolutions will be documented in the CESSAR-DC draft Safety Evaluation Reports.

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
002	FAILURE OF PROTECTIVE DEVICES ON ESSENTIAL EQUIPMENT	GSI	TO BE REP.	7.4
003	SETPOINT DRIFT IN INSTRUMENTATION	GSI	NEARLY RES	7.1.2
012	BWR JET PUMP INTEGRITY	GSI	MEDIUM	NA
014	PWR PIPE CRACKS	GSI	NEARLY RES	5.2.3
020	EFFECTS OF ELECTROMAGNETIC PULSE ON NUCLEAR PLANT SYSTEMS	GSI	NEARLY RES	7.1
022	INADVERTANT BORON DILUTION EVENTS	GSI	NEARLY RES	5.4.6
023	REACTOR COOLANT PUMP SEAL FAILURES	GSI	HIGH	5.4.1
024	AUTOMATIC EMERGENCY CORE COOLING SYSTEM SWITCH TO RECIRCULATION	GSI	TO BE REP.	6.3
029	BOLTING DEGRADATION OR FAILURES IN NUCLEAR PLANTS	GSI	HIGH	5.2.3
036	LOSS OF SERVICE WATER	GSI	NEARLY RES	9.2.1
038	POTENTIAL RECIRCULATION FAILURE AS A CONSEQUENCE OF CONTAINMENT PAINT OR DEBRIS	GSI	TO BE DET.	6.1.2
040	SAFETY CONCERNS ASSOC. WITH BREAKS IN THE BWR SCRAM SYSTEM	GSI	NEARLY RES	NA
045	INOPERABILITY OF INSTRUMENTS DUE TO EXTREME COLD WEATHER	GSI	NEARLY RES	7.1

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
048	LCO FOR CLASS 1E VITAL INSTRUMENT BUSES IN OPERATING REACTORS	GSI	NEARLY RES	16
049	INTERLOCKS AND LCO'S FOR REDUNDANT CLASS 1E TIE BREAKER	GSI	MEDIUM	7 OR 8
050	REACTOR VESSEL LEVEL IN BWRS	GSI	NEARLY RES	NA
051	PROP.REQ.FOR IMPROVING REL.OF OPEN CYCLE SER.WTR	GSI	MEDIUM	9
055	FAILURE OF CLASS 1E SAFETY RELATED SWITCHGEAR CIRCUIT BREAKER TO CLOSE ON DEMAND	GSI	TO BE REP.	8.3
057	EFFECTS OF FIRE PROTECTION SYSTEM ACTUATION ON SAFETY RELATED EQUIPMENT	GSI	TO BE DET.	7 OR 9
061	SRV DISCHARGE LINE BREAK INSIDE TO WETWELL AIRSPACE OF MARK I & III CONTAINMENT	GSI	MEDIUM	NA
062	REACTOR SYSTEMS BOLTING APPLICATIONS	GSI	TO BE DET.	5.2.3
063	USE OF EQUIPMENT NOT CLASSIFIED AS ESSENTIAL TO SAFETY IN BWR TRANSIENT ANALYSIS	GSI	TO BE DET.	NA
065	PROBABILITY FO CORE MELT DUE TO COMPONENT COOLING WATER SYSTEM FAILURES	GSI	HIGH	PRA APPDX
066	STEAM GENERATOR REQUIREMENTS	GSI	NEARLY RES	5.4.2

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
067	STEAM GENERATOR STAFF ACTIONS	GSI	MEDIUM	5.2
067.7	STEAM GENERATOR STAFF ACTIONS-EDDY CURRENT TESTS	GSI	MEDIUM	
068	POSTULATED LOSS OF AFWS RESULTING FROM TURBINE DRIVEN AFW PUMP STEAM SUPPLY LINE BREAK	GSI	HIGH	10.4
069	MAKE-UP NOZZLE CRACKING IN B&W PLANTS	GSI	NEARLY RES	NA
070	PORV AND BLOCK VALVE RELIABILITY	GSI	MEDIUM	5
071	FAILURE OF RESIN DEMINERALIZER SYSTEMS AND THEIR EFFECTS ON PLANT SAFETY	GSI	TO BE DET.	9.3
072	CONTROL ROD DRIVE GUIDE TUBE SUPPORT PIN FAILURES	GSI	TO BE DET.	4.5.2
073	DETACHED THERMAL SLEEVES	GSI	TO BE DET.	5.4.3
075	GEN. IMPLICATIONS OF ATWS EVENTS AT SALEM	GSI	NEARLY RES	7.1
077	FLOODING OF SAFETY EQUIPMENT COMPARTMENTS BY BACKFLOW	GSI	HIGH	6
078	MONITORING OF FATIGUE TRANSIENT LIMITS FOR REACTOR COOLANT SYSTEM	GSI	TO BE DET.	5.2
079	UNANALYZED REACTOR VESSEL THERMAL STRESS-COOLDOWN	GSI	MEDIUM	5.3.2

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
082	BEYOND DESIGN BASES ACCIDENTS IN SPENT FUEL POOLS	GSI	MEDIUM	9.1
083	CONTROL ROOM HABITABILITY	GSI	NEARLY RES	18
084	CE PORVS	GSI	NEARLY RES	5.2
086	LONG RANGE PLAN FOR DEALING W/SSC IN BWR PIPING	GSI	NEARLY RES	NA
087	FAILURE OF HPCI STEAM LINE WITHOUT ISOLATION	GSI	HIGH	NA
088	EARTHQUAKE AND EMERGENCY PLANNING	GSI	TO BE DET.	2 & 13
089	STIFF PIPE CLAMPS	GSI	TO BE DET.	5.4.3
091	MAIN CRANKSHAFT FAILURE IN TRANSAMERICA DELAVAL EDG'S	GSI	NEARLY RES	8
093	STEAM BINDING OF AUXILIARY FEEDWATER PUMPS	GSI	HIGH	10.9.4
094	ADDITIONAL LTOP FOR LIGHT WATER REACTORS	GSI	HIGH	5.3.2
095	LOSS OF EFFECTIVE VOLUME FOR CONTAINMENT RECIRCULATION	GSI	TO BE DET.	6.2.2
096	RHR SUCTION VALVE TESTING	GSI	TO BE DET.	5.4.7
099	RCS/RHR SUCTION LINE INTERLOCKS ON PWRS	GSI	HIGH	5.4.7
100	OTSG LEVEL	GSI	TO BE DET.	NA
101	BWR WATER LEVEL REDUNDANCY	GSI	HIGH	NA

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
102	HUMAN ERROR IN EVENTS INVOLVING WRONG UNIT OR WRONG TRAIN	GSI	NEARLY RES	18
103	DESIGN FOR PROBABLE MAXIMUM PRECIPATATION	GSI	NEARLY RES	2
104	REDUCTION OF BORON DILUTION REQUIREMENTS	GSI	TO BE DET.	15.4.6
105	INTERFACING SYSTEMS LOCA AT BWRS	GSI	HIGH	NA
106	PIPING AND USE OF HIGHLY COMBUSTIBLE GASES IN VITAL AREAS	GSI	TO BE DET.	6
107	GENERIC IMPLICATIONS OF MAIN TRANSFORMER FAILURES	GSI	TO BE DET.	8
109	REACTOR VESSEL CLOSURE FAILURE	GSI	TO BE DET.	15.6
110	EQUIPMENT PROTECTION DEVICES ON ENGINEERED SAFETY FEATURES	GSI	TO BE DET.	6.0
113	QUALIFICATION TESTING OF LARGE BORE HYDRAULIC SNUBBERS	GSI	TO BE DET.	3.9.2
115	ENHANCEMENT OF THE RELIABILITY OF THE WEST. SSPS	GSI	HIGH	7
116	ACCIDENT MANAGEMENT	GSI	TO BE DET.	18
117	ALLOWABE OUTAGE TIMES FOR DIVERSE SIMULTANEOUS EQUIPMENT OUTAGES	GSI	TO BE DET.	7 OR 16
118	TENDON ANCHORAGE FAILURE	GSI	TO BE DET.	3.8

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
120	ON-LINE TESTABILITY OF PROTECTION SYSTEMS	GSI	TO BE DET.	3.9.6
121	HYDROGEN CONTROL FOR LARGE DRY PWR CONTAINMENTS	GSI	HIGH	6.2
122.1A	COMMON MODE FAILURE OF ISOLATION VALVES IN CLOSED POSITIONS	GSI	HIGH	10.4.9
122.1B	RECOVERY OF AUXILIARY FEEDWATER	GSI	MEDIUM	10.4.9
122.1C	INTERRUPTION OF AUXILIARY FEEDWATER FLOW	GSI	HIGH	10.4.9
122.2	INITIATING FEED AND BLEED	GSI	HIGH	9
123	DEFERRMENT IN THE REGULATIONS GOVERNING DBA AND SINGLE FAILURE CRITERION - DAVIS BESSE	GSI	TO BE DET.	10.4
124	AUXILIARY FEEDWATER SYSTEM RELIABILITY	GSI	NEARLY RES	10.4.9
125.I.3	SPDS AVAILABILITY	GSI	TO BE DET.	18.2
125.I.5	SAFETY SYSTEM TESTED IN ALL CONDITIONS REQUIRED BY DESIGN BASIS ANALYSIS	GSI	TO BE DET.	7 OR 16
125.I.6	VALVE TORQUE LIMIT AND BYPASS SWITCH SETTINGS	GSI	TO BE DET.	3.9.6
125.I.7.a	RECOVER FAILED EQUIPMENT	GSI	TO BE DET.	TO BE DET
125.I.8	PROCEDURES AND STAFFING FOR REPORTING TO NRC EMERGENCY RESPONSE CENTER	GSI	TO BE DET.	NA

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
125.II.1.b	REVIEW EXISTING AFWS FOR SINGLE FAILURE	GSI	HIGH	10
125.II.11	RECOVERY OF MAIN FEEDWATER AS AN ALTERNATIVE TO AFW	GSI	TO BE DET.	10.4
125.II.13	OPERATOR JOB AIDS	GSI	TO BE DET.	13
125.II.2	ADEQUACY OF EXISTING MAINTENANCE REQUIREMENTS FOR SAFETY RELATED SYSTEMS	GSI	TO BE DET.	5
125.II.5	THERMAL-HYDRAULIC EFFECTS-LOSS AND RESTORATION OF FDW ON PRIMARY SYSTEM COMPONENTS	GSI	??	NA
125.II.7	REEVALUATE AUTO ISO OF FDW FROM SG DURING LINE BRK	GSI	HIGH	15.2
126	RELIABILITY OF PWR MAIN STEAM SAFETY VALVES	GSI	TO BE DET.	5.2
127	TESTING AND MAINTENANCE OF MANUAL VALVES IN SAFETY RELATED SYSTEMS	GSI	TO BE DET.	3.9.6
128	ELECTRICAL POWER RELIABILITY	GSI	HIGH	8 OR PRA APPDX.
129	VALVE INTERLOCKS TO PREVENT VESSEL DRAINAGE DURING SHUTDOWN COOLING	GSI	TO BE DET.	5.4.7
130	ESSENTIAL SERVICE WATER PUMP FAILURES AT MULTIPLANT SITES	GSI	HIGH	6
131	POTENTIAL SEISMIC INTERACTION INVOLVING THE MOVABLE INCORE FLUX MAP SYSTEM AT WESTINGHOUSE PLANTS	GSI	TO BE DET.	NA

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
132	RHR PUMPS INSIDE CONTAIMENT	GSI	TO BE DET.	5.4.7
134	DEGREE AND EXPERIENCE REQ. FOR SENIOR OPERATORS	GSI	HIGH	NA
135	INTEGRATED STEAM GENERATOR ISSUES	GSI	TO BE DET.	5.4.2
136	STORAGE AND USE OF LARGE QUANTITIES OF CRYOGENIC COMBUSTIBLES	GSI	TO BE DET.	6
137	REFUELING CAVITY SEAL FAILURES	GSI	TO BE DET.	9.1.4
138	DEINERTING UPON DISCOVERY OF RCS LEAKAGE	GSI	TO BE DET.	5 & 16
139	THINNING OF CARBON STEEL PIPING IN LWRS	GSI	TO BE DET.	3 OR 5 OR 10
140	FISSION PRODUCT REMOVAL BY CONTAINMENT SPRAYS OR POOLS	GSI	TO BE DET.	6
A-01	WATER HAMMER	USI	USI	5.4.2
A-02	ASYMETRIC BLOWDOWN LOADS ON RCS	USI	USI	15.6
A-03	WESTINGHOUSE STEAM GENERATOR TUBE INTEGRITY	USI	USI	5
A-04	C-E STEAM GENERATOR TUBE INTEGRITY	USI	USI	5.4.2
A-05	B&W STEAM GENERATOR TUBE INTEGRITY	USI	USI	NA
A-09	ATWS	USI	USI	15.8

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
A-10	BWR FEEDWATER NOZZLE CRACKING	USI	USI	NA
A-11	REACTOR VESSEL MATERIAL TOUGHNESS	USI	USI	5.3
A-12	FRACTURE TOUGHNESS OF S.G. AND RCP SUPPORTS	USI	USI	5.4.14
A-17	SYSTEMS INTERACTION	USI	USI	5 & 15
A-19	DIGITAL COMPUTER PROTECTION SYSTEM	GSI	TO BE REP.	7.2
A-24	QUALIFICATION OF CLASS 1E SAFETY RELATED EQUIPMENT	USI	USI	7.1.2.5
A-26	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION	USI	USI	5.3.2
A-29	PLANT DESIGN FOR REDUCT. OF VULNER. TO SABOTAGE	GSI	MEDIUM	2
A-31	RHR SHUTDOWN REQUIREMENTS	USI	USI	5.4.7
A-36	CONTRCL OF HEAVY LOADS NEAR SPENT FUEL	USI	USI	9
A-39	DETERMINATION OF SAFETY RELIEF VLV POOL DYN LOADS	USI	USI	NA
A-40	SEISMIC DESIGN--SHORT TERM PROGRAM	USI	USI	3.7
A-41	LONG TERM SEISMIC PROGRAM	USI	USI	3.7
A-42	PIPE CRACKS IN BOILING WATER REACTORS	USI	USI	NA
A-43	CONTAINMENT EMERGENCY SUMP PERFORMANCE	USI	USI	6.2

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
A-44	STATION BLACKOUT	USI	USI	15.3
A-45	SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS	USI	USI	5.4.7
A-46	SEISMIC QUAL. OF EQUIPMENT IN OPERATING PLANTS	USI	USI	NA
A-47	SAFETY IMPLICATIONS OF CONTROL SYSTEMS	USI	USI	5.3
A-48	HYDROGEN CNTRL MEASURES&EFFECTS OF HYDROGEN BURNS	USI	USI	6.2
A-49	PRESSURIZED THERMAL SHOCK	USI	USI	7.1
B-05	DUCTILITY OF TWO-WAY SLABS AND SHELLS -STEEL CONTM	GSI	MEDIUM	3.8
B-06	LOAD, LOAD COMBINATIONS, STRESS LIMITS	GSI	HIGH	3.9.3
B-10	BEHAVIOR OF BWR MARK III CONTAINMENTS	GSI	HIGH	NA
B-17	CRITERIA FOR SAFETY RELATED ACTIONS	GSI	HIGH	5 OR 13 O 18
B-19	THERMAL-HYDRAULIC STABILITY	GSI	NEARLY RES	4.4
B-22	LWR FUEL	GSI	TO BE REP.	4.2
B-26	STRUCTURAL INTEGRITY OF CONTAINMENT PENETRATIONS	GSI	MEDIUM	6.2
B-29	EFFECTIVENESS OF ULTIMATE HEAT SINKS	GSI	TO BE REP.	6 OR 9

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
B-31	DAM FAILURE MODEL	GSI	TO BE REP.	2.4.2
B-32	ICE EFFECTS ON SAFETY RELATED WATER SUPPLIES	GSI	TO BE REP.	9
B-53	LOAD BREAK SWITCH	GSI	NEARLY RES	8
B-54	ICE CONDENSER CONTAINMENTS	GSI	MEDIUM	NA
B-55	IMPROVE RELIABILITY OF TARGET ROCK SAFETY RELIEF VALVES	GSI	MEDIUM	5
B-56	DIESEL GENERATOR RELIABILITY	GSI	HIGH	8 & PRA APPDX.
B-58	PASSIVE MECHANICAL FAILURES	GSI	MEDIUM	3 OR 15 O PRA
B-60	LOOSE PARTS MONITORING SYSTEM	GSI	NEARLY RES	5
B-61	ALLOWABLE ECCS EQUIPMENT OUTAGE PERIODS	GSI	MEDIUM	6 OR 16
B-64	DECOMMISSIONING OF REACTORS	GSI	NEARLY RES	NA
C-08	MAIN STEAM LINE ISOLATION VALVE LEAKAGE CNTRL SYS.	GSI	HIGH	NA
C-09	RHR HEAT EXCHANGER TUBE FAILURES	GSI	TO BE REP.	5.4.7
C-11	ASSESSMENT OF FAILURE AND RELIABILITY OF PUMPS AND VALVES	GSI	MEDIUM	3.9.6
C-14	STORM SURGE MODES FOR COASTAL SITES	GSI	TO BE DET.	NA

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
D-02	EMERGENCY CORE COOLING SYSTEM CAPABILITY FOR FUTURE PLANTS	GSI	TO BE REP.	6.3
HF 1	HUMAN FACTORS PROGRAM PLAN	GSI	HIGH	18
HF 1.1	SHIFT STAFFING	GSI	HIGH	13
HF 1.2	ENGINEERING EXPERTISE ON SHIFT	GSI	HIGH	13
HF 1.3	GUIDANCE ON LIMITS AND CONDITIONS OF SHIFT WORK	GSI	HIGH	13
HF 2	MAINTENANCE AND SURVEILLANCE PROGRAM PLAN	GSI	HIGH	18
HF 4.1	INSPECTION PROCEDURE FOR UPGRADING EMER. OP. PROC.	GSI	HIGH	13
HF 4.4	GUIDELINES FOR UPGRADING OTHER PROCEDURES	GSI	HIGH	
HF 5.1	LOCAL CONTROL STATIONS	GSI	HIGH	18
HF 5.2	REVIEW CRITERIA FOR HF ASPECTS OF ADVANCED I&C	GSI	HIGH	18
HF 8	MAINTENANCE AND SURVEILLANCE PROGRAM	GSI	HIGH	18
I.A.1.4	LONG TERM UPGRADING OF OPERATING PERSONNEL	GSI	NEARLY RES	NA
I.A.2.2	TRAINING AND QUALIFICATIONS OF OPERATING PERSONNEL	GSI	HIGH	13 OR 18
I.A.2.6(1)	REVISE REGULATORY GUIDE 1.8	GSI	HIGH	NA

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
I.A.2.6(4)	OPERATOR WORKSHOPS	GSI	MEDIUM	NA
I.A.2.7	ACCREDITATION OF TRAINING INSTITUTIONS	GSI	MEDIUM	NA
I.A.3.3	REQUIREMENTS FOR OPERATOR FITNESS	GSI	HIGH	NA
I.A.3.4	LICENSING OF ADDITONAL OPERATOR PERSONNEL	GSI	MEDIUM	NA
I.A.4.2	RESEARCH ON TRAINING SIMULATORS	GSI	HIGH	NA
I.B.1	(1-4) ORGANIZATION AND MANAGEMENT - LONG TERM IMPROVEMENTS	GSI	MEDIUM	13
I.C.9	LONG TERM PLAN FOR UPGRADING OF PROCEDURES	GSI	MEDIUM	13
I.D.3	SAFETY SYSTEM STATUS MONITORING	GSI	MEDIUM	18.2
I.D.4	CONTROL ROOM DESIGN STANDARD	GSI	MEDIUM	18.1
I.D.5(3)	ON-LINE REACTOR SURVEILLANCE SYSTEMS	GSI	NEARLY RES	7 OR 18
I.D.5(5)	DISTURBANCE ANALYSIS SYSTEMS	GSI	MEDIUM	18.1
I.F.1	QUALITY ASSURANCE - EXPAND QUALITY ASSURANCE LIST	GSI	HIGH	17.1
I.G.2	SCOPE OF TEST PROGRAM	GSI	MEDIUM	14.2
II.A.1	SITING POLICY REFORMULATION	GSI	MEDIUM	2

UNRESOLVED SAFETY ISSUES & HIGH/MEDIUM PRIORITY GENERIC ISSUES WHICH WERE EVALUATED FOR APPLICABILITY TO SYSTEM 80+ DESIGN CERTIFICATION

ISSUE NO.	ISSUE TITLE	ISSUE TYPE	ISSUE PRIORITY AND STATUS	CESSAR-DC SECTION
II.B.5	EFFECT OF H2 BURNING AND EXPLOSIONS ON CONT STRUCT	GSI	MEDIUM	6.2
II.B.6	RISK REDUCTION FOR OPERATING REACTORS WITH SITES WITH HIGH POPULATION DENSITIES	GSI	HIGH	NA
II.B.8	RULEMAKING PROCEEDINGS ON DEGRADED CORE ACCIDENTS-HYDROGEN RULE, SEVERE ACCEDENT, ETC.	GSI	HIGH	6.2.5
II.C.1	INTERIM RELIABILITY EVALUATION PROGRAM	GSI	HIGH	NA
II.C.2	CONTINUATION OF INTERIM RELIABILITY EVALUATION PROGRAM	GSI	HIGH	NA
II.C.4	RELIABILITY ENGINEERING	TMI	HIGH	NA
II.E.2.2	RESEARCH ON SMALL BREAK LOCAs AND ANOMALOUS TRANSIENTS	GSI	MEDIUM	6 & 15.6
II.E.4.3	CONTAINMENT INTEGRITY CHECK	TMI	HIGH	6 & 16
II.E.5.2	B&W REACTOR TRANSIENT RESPONSE TASK FORCE	GSI	NEARLY RES	NA
II.E.6.1	TEST ADEQUACY STUDY	GSI	MEDIUM	5 & 16
II.F.5	CLASSIFICATION OF I & C, AND ELECTRICAL EQUIPMENT	GSI	MEDIUM	7.1
II.H.2	OBTAIN DATA ON INSIDE COND. OF TMI CONTAINMENT	TMI	HIGH	6
II.J.4.1	REVISE DEFICIENCY REPORT REQUIREMENTS	GSI	NEARLY RES	NA

APPENDIX B

Combustion Engineering Design Certification Program

Process for Probabilistic Risk
Assessment as Required by the
Severe Accident Policy Statement

I. Overview of Process for Probabilistic Risk Assessment of System 80+

One of the requirements of the NRC's Severe Accident Policy Statement is that a Probabilistic Risk Assessment (PRA) must be performed for all future plants. To address these requirements, a System 80+ Standard Design Level III PRA is being performed as part of the DOE ALWR Design Verification Program.

The System 80+ Standard Design PRA has two primary purposes. The first purpose is to identify (1) the dominant contributors to severe accident risk and (2) the accident sequences which are insignificant. The second purpose is to provide an analytical tool for evaluating the impact of design modifications on core damage probability and risk to the health and safety of the public.

This PRA is being performed in two phases. In the first phase, Event Trees and Fault Tree Models are being developed for the current System 80 design. These models will be used to establish a baseline core damage frequency for the current System 80 design and to determine the dominant core damage contributors for System 80.

The second phase will be an interactive process in which these models will be modified to reflect system design changes proposed for System 80+. The models will be reevaluated to determine the impact of the design changes on core damage frequency and dominant core damage contributors. These impacts will be reviewed and additional design changes will be considered as appropriate to achieve the risk reduction requirements.

Phase One: Baseline System 80 PRA

The baseline System 80 core damage frequency calculation performed for the DOE ALWR Design Verification Program is a Level 1 PRA for the System 80 Nuclear Steam Supply System (NSSS) described in CESSAR-F. This PRA includes the identification and quantification of accident sequences attributable to internal initiators which lead to core damage. While the Balance of Plant (BOP) systems are outside of the System 80 NSSS scope, information on certain BOP systems is required in order to thoroughly evaluate the performance of the NSSS Systems. Where such information is required, functional system designs which meet CESSAR-F interface requirements and are consistent with support system configurations used in recent vintage C-E plants will be used in the analyses.

Phase Two: System 80+ PRA

As the System 80 design evolves into the System 80+ Standard Design (the Nuclear Power Module and Standardized Functional Descriptions), the baseline PRA will also evolve so as to provide input to the many design decisions that will be made. Based on the results of the Baseline PRA, initial system reliability targets will be established and potential system weak links will be identified.

Recognizing that some system reliability targets will be more difficult or expensive to meet than others, trade-offs will be called for and the evolving PRA will serve as an "accounting" tool to monitor the current status of the design with respect to reliability and risk goals. These goals include reliability goals from Standard Review Plans, large-release frequency goals from the Safety Goal Policy Statement and EPRI ALWR Program core melt frequency objectives.

The baseline PRA will identify dominant accident sequences with occurrence frequencies high enough to preclude meeting the goals. System 80+ Standard Design development efforts will then be focused on improving the reliability of systems or equipment involved in the dominant sequences. As design improvements are adopted, the PRA models will be updated so as to provide a current list of dominant sequences.

The final PRA for the System 80+ Standard Design will consist of the baseline PRA updated to include all of the design modifications that are implemented as a part of the ALWR Design Verification Program. Additionally, with support from the DOE Advanced Reactor Severe Accident Program (ARSAP), the PRA will be upgraded to a Level III PRA and External Events will be addressed generically.

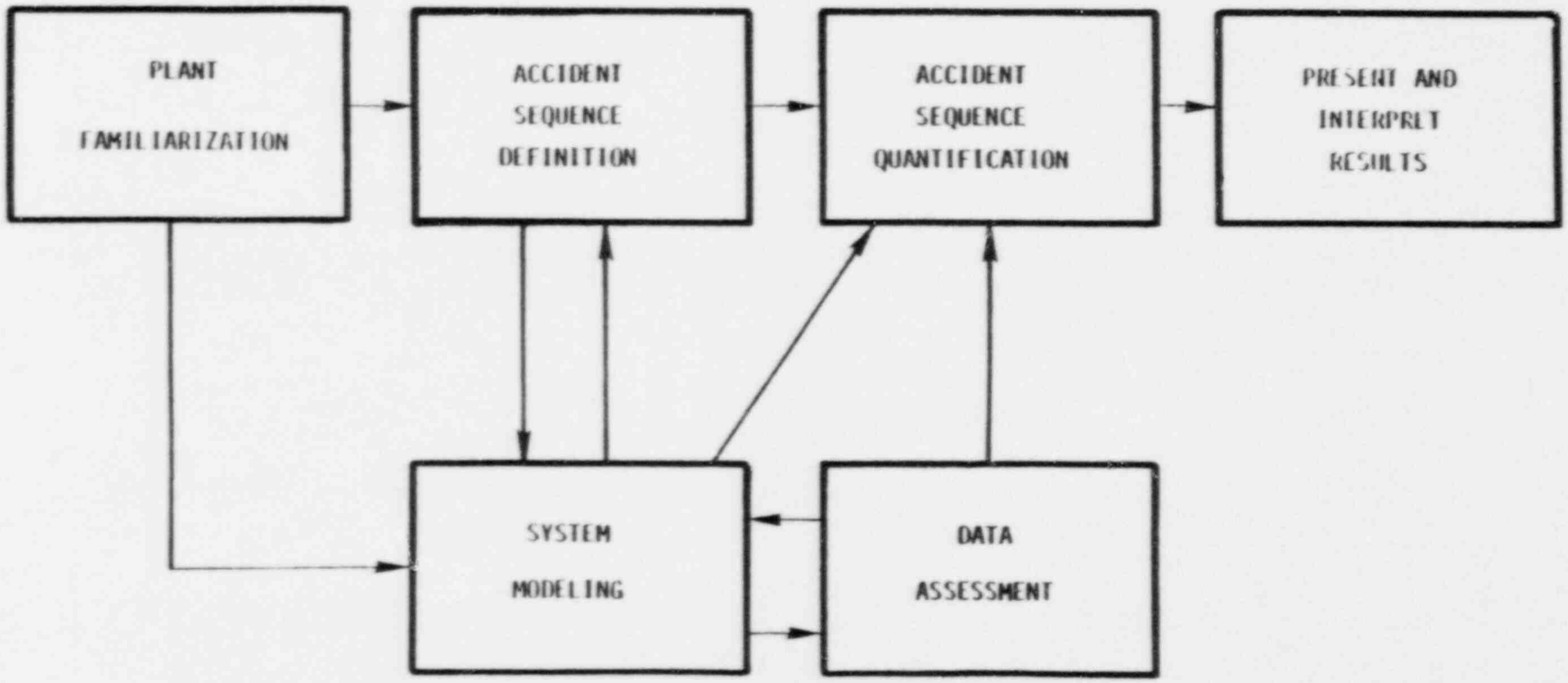
II. Acceptance Criteria and Methodology for PRA

As stated in Section I, the objectives of PRA analyses are to calculate a baseline core damage frequency for a generic System 80 plant, to determine the dominant core damage contributors and to assess potential areas for design improvements in the System 80+ Standard Design and to document the System 80+ Standard Design PRA. These analyses are equivalent to the Probabilistic Safety Analysis

(PSA) described in the PSA Procedures Guide (NUREG/CR-2815). The methods employed in this analysis are consistent with methods outlined in the PSA Procedures Guide and methods described in the PRA Procedures Guide (NUREG/CR-2300). This work will use the small event tree/large fault tree approach. Figure B-1 shows the major tasks in this analysis. The following sections describe each of these tasks and associated methodology.

FIGURE B-1

MAJOR PRA TASKS



Plant Familiarization

The objective of this task is to (1) collect the information necessary for identification of appropriate initiating events, (2) determine the success criteria for the front line systems required to prevent or mitigate the transients and accidents and (3) identify the dependence between the front line systems and the support systems which are required for proper functioning of the front line systems. This task is primarily an information gathering task.

The information collected in this task includes design information, operational information and information on plant responses to transients. CESSAR-F will be used to provide information on the design of systems within the basic NSSS scope and interface requirements for the support systems. Where additional design detail is needed for support systems, typical system designs will be generated based on support system designs described in the FSARs of recent vintage C-E plants with similar NSSS designs.

Operator actions during plant transients will be evaluated and established based on C-E's Emergency Procedure Guidelines and discussions with licensed operators in C-E's Training Department and at an operating System 80 plant. Surveillance requirements and operability definitions will be derived from C-E's Standard Technical Specifications and, where more specific detail is needed, from System 80 plant specific Technical Specifications. Maintenance information, where needed, will be based on common industry practices.

The Reactor Safety Study, several other published PRA studies, and the IDCOR IPE Procedures Guide will also be reviewed as part of the plant familiarization task. The objectives of these reviews are to provide a broad overview of areas to be addressed in this analysis and to identify potential problem areas.

Accident Sequence Definition

The objective of this task is to qualitatively identify those accident sequences which lead to core melt/core damage. This will be accomplished using event tree analysis. Event tree analysis involves defining a set of initiating events and constructing a set of system event trees which relate plant system responses to each defined initiating event. Each system event tree represents a distinct set of system accident sequences, each of which consists of an initiating event and a combination of various system successes and failures that lead to an identifiable plant state. Procedures for developing system event trees are described in detail in the PRA Procedures Guide. For this analysis, the small event tree/large fault tree approach will be used. In this approach, only the front line systems which respond to mitigate an accident or transient, will be addressed on the event tree. The impact of the support systems is addressed within the fault tree models for the front line systems.

A Master Logic Diagram (MLD) will be constructed to guide the selection and grouping of the initiating events. An MLD is essentially a top level tree in which the general conditions that could lead to the top level event are deductively determined. For this analysis, the top event on the MLD is defined to be "offsite release" even though the scope of the analysis is limited to identifying core damage frequency and dominant contributors. This is to ensure completeness and to facilitate later extension of this analysis.

System Modeling

Quantification of the system accident sequences requires knowledge of the failure probability or frequency of occurrence for each element of the system accident sequence. The initiating event frequency and the probability of failure for a system accident sequence element involving the failure of a single component can be quantified directly from the appropriate raw data. However, if the system accident sequence element represents a specific failure mode for a system or subsystem, a fault tree model of the system or subsystem will be constructed and quantified to obtain the desired failure probability.

The evaluation of each fault tree yields both qualitative and quantitative information. The quantitative evaluation of the fault trees yields several numerical measures of a systems failure probability, two of which are typically employed in the event tree quantification (i.e., the unavailability and unreliability).

The unavailability is the probability that a system will not respond when demanded. The unreliability is the probability that a system will fail (at least once) during a given required operating period. The unreliability is usually added to the unavailability when the system accident sequence element represents the failure of a standby system to actuate and then run for a specified period of time.

Two types of human failures will be included in the fault tree analyses. They are "pre-existing maintenance errors" and failures of the operator to respond to various demands. Pre-existing maintenance errors are undetected errors committed since the last periodic test of a standby system. An example of this type of error is the failure to reopen a mini-flow valve which was closed for maintenance. A failure of the operator to respond includes the failure of the operator to perform a required function at all or to perform it correctly. An example of this type of error is the failure of the operator to back-up the automatic actuation of a safety system.

For this PRA, failure of the operator to respond to various demands where there was a time constraint will be quantified using the Human Cognitive Reliability Model. The human cognitive reliability model is a set of time dependent functions which describe the probability of a crew response in performing a task. The human cognitive reliability model permits the analyst to predict the cognitive reliability associated with a non-response for a given task or series of related tasks, once the dominant type of cognitive processing (skill-based, rule-based or knowledge-based), the medium response time for the task or tasks under nominal conditions and performance shaping factors such as stress levels or environment are identified. The inherent time dependence in this model makes it ideal for evaluating operator responses during a transient. The failure probability for "pre-existing maintenance errors" will be quantified using the Handbook of Human Reliability Analysis. The Handbook of Human Reliability Analysis is an extension of the human reliability analysis methodology developed for WASH 1400, the Reactor Safety Study, and is intended to provide methods, models and estimated human error probabilities to enable analysts to make quantitative or qualitative assessments of the occurrence of human errors that affect the availability or operational reliability of engineered safety systems and components. The emphasis is on tasks addressed in the Reactor Safety Study, calibration, maintenance and selected control room tasks related to engineered safety features availability. It is the best available source for evaluating human performance with respect to maintenance, calibration, testing and other tasks performed during normal plant operation. However, the time dependent model is not as thorough and explicit as that provided by the human cognitive reliability model.

For this PRA, the small event tree/large fault tree approach has been selected. The event trees developed for this PRA will address the response of the front line systems, that is, those systems directly involved in mitigating the various initiating events. The impact of the support systems will be modeled within the front line system models. CESSAR-F contains interface requirements for the support systems but does not contain any support system configurations or schematics. Therefore, in order to develop the support system models, representative support system configurations will be developed using the CESSAR-F interface requirements, support system configurations for System 80 plants and the typical system configurations in the Nuclear Plant Reliability Data System (NPRDS) Reportable Scope Manual for C-E Plants.

Once the baseline PRA models are established, they will be used in the reliability assurance program mentioned above. The models will identify where improvements are needed to assure reliability, risk, and core melt frequency goals are met. If system designs evolve, for example, from two-train to four-train systems, the system models will be revised in order to provide an up-to-date assessment of where the design stands compared to the goals and to identify potential areas for improvement. As the Standardized Functional Descriptions are developed for CESSAR-DC, and as additional requirements from the EPRI ALWR Requirements Document are adopted, the system models will be updated to reflect those requirements. The System Reliability Models that result from this process will form the heart of the final System 80+ Standard Design PRA.

Data Assessment

Reliability data is needed for the quantification of the system fault trees and the system accident sequences which result in severe core damage. The data needed for this quantification includes:

1. initiating event frequencies,

2. component failure rates (demand and time-dependent),

3. component repair times and maintenance frequencies,

4. common cause failure rates,

5. human failure probabilities,

6. special event probabilities (e.g., restoration of offsite power), and

7. error factors for the items above.

Because the analysis is for a generic System 80 plant, generic reliability data will be used in this analysis. The basic initiating event frequencies will be extracted from the PSA Procedure Guide, EPRI NP-2230 and the NREP Generic Data Base. The initiating event frequencies in the Zion PRA, the Oconee PRA and Calvert Cliffs IREP Report will also be considered.

Accident Sequence Quantification

The basic objective of this analysis is to model baseline core damage frequency for a generic System 80 plant and then again for the System 80+ Standard Design. The total core damage frequency, due to internal events, is the sum of the frequencies of the system level accident sequence frequencies for those accident sequences which result in core damage.

The system level accident sequences leading to core damage will be identified using event tree analysis. Each system level accident sequence will consist of an initiating event and one or more additional elements, each representing either a front line system failure or a special event such as failure to restore off site power within a given time or the most reactive rod sticking out of the core. The frequency for the system level accident sequence will be determined by quantifying the individual elements in the sequence and then combining the results in the appropriate manner. The frequencies for the initiating events and the special events are directly calculable.

The front line system failure probabilities will be calculated in the baseline analysis using conditioned fault tree analysis. In the System 80+ Standard Design PRA, fault tree linking will be used. The first step in this process will be to construct a fault tree model for each front line system that appeared as an element in a system accident sequence. The models will include submodels for the appropriate support systems.

The next step will be to perform a baseline quantification of each fault tree using generic failure rates. For those front line systems appearing in the LOCA or steam line break sequences, base line quantifications will be made with and without offsite power. This quantification provides a list of cutsets, the system unreliability and the system unavailability for each front line system. This quantification will be performed using CEREC, a fault tree analysis computer code. The third step in this process is to identify common elements in fault tree models appearing in any given event sequence and to calculate conditional failure probabilities for these elements.

After all the conditioned component failure rates are calculated, the system fault trees will be requantified using the appropriate conditioned component failure rates, thus yielding a set of system failure probabilities specific to the initiating event classes.

The final step in the quantification of the core damage frequency is to solve each system accident sequence equation using the appropriate initiating event, special event and system failure probabilities. This will be done using CESAM, a Monte Carlo sampling code for equation solving.

Radionuclide Release and Transport

The evaluation of environmental radionuclide release that result from severely degraded core accidents will involve four elements:

1. Radionuclide and structural material inventories;
2. Radionuclide and structural material source term from the core;
3 Transport, deposition, and release in the primary system; and
4. Transport, deposition, and release in the containment.

The analysis will proceed in a sequential manner, starting with the radionuclide and structural material inventories. This will involve the determination of the quantities of radionuclides and structural materials that are present at the beginning of an accident. The next step will be the evaluation o. the radionuclide and structural material source term from the core. This will entail the determination of the quantities of radionuclides and structural materials released from the core to the primary system or to the containment. (Direct releases of radionuclides and structural materials from the corium--the melted core and structural materials--to the containment can occur in meltdown accidents after the pressure vessel has melted through and the corium is interacting with the concrete basemat.) This source term will then be used in the analysis of radionuclide transport, deposition, and release in the primary system. The analysis will consider the various deposition processes that can occur in the primary system. The result will be the source term for release from the primary system to the containment; it is used in the analysis of transport, deposition, and release in the containment. This analysis will take account of the various deposition processes that can occur in the containment, and it will determine the quantities of radionuclides released from the containment to the environment.

III. NRC Review Process and Documentation

The System 80+ Standard Design Probabilistic Risk Assessment will be documented in an appendix to CESSAR-DC and submitted to the NRC in June 1989. In the meantime, however, Combustion Engineering will apprise the NRC and obtain NRC feedback on the System 80+ Standard Design PRA via meetings and draft reports. The purpose of these early interactions is to provide continuous NRC comments as the System 80+ Standard Design PRA is developed. Emphasis will be placed on establishing NRC criteria for acceptance of the System 80+ PRA. These comments and preliminary criteria will be documented in meeting minutes issued by NRC.

Combustion Engineering will document, in the CESSAR-DC appendix all acceptance criteria and descriptive information necessary to obtain NRC concurrence on the System 80+ Standard Design PRA. NRC concurrence on the CESSAR-DC PRA appendix will be provided in the Safety Evaluation Report.

APPENDIX C

Combustion Engineering Design Certification Program

Process for Degraded Core Evaluation as Required by the Severe Accident Policy Statement.

I. Overview of Process for Degraded Core Evaluation

The NRC Severe Accident Policy Statement (SAPS) requires that the design bases for future plants include consideration of both the prevention and mitigation of degraded core accidents, using an evaluation approach based on deterministic engineering analysis and judgement, complemented by Probabilistic Risk Assessment (PRA). Combustion Engineering, with support by the DOE Advanced Reactor Severe Accident Program (ARSAP), will include degraded core evaluation in the design of the System 80+ Standard Design (the Nuclear Power Module and Standardized Functional Descriptions). The proposed approach for this evaluation is to identify the severe accident issues applicable to the System 80+ Standard Design, to develop criteria for resolution of those issues, and to develop the method of resolution of each issue for the System 80+ Standard Design. Completion of the review of this evaluation (in support of the System 80+ Design Certification) will require NRC approval of (1) the completeness and applicability of the list of issues identified, (2) the criteria for resolution of the severe accident issues in this list, and (3) the method of resolution of the issues in this list.

II. Method of Evaluation

ARSAP has identified severe accident issues on the basis of results of the Industry Degraded Core Rulemaking (IDCOR) Program and current research related to severe accidents. These issues will be addressed in Topic Papers which document technical information on the subject issues and propose criteria for resolution of those issues. The resolution of severe accident issues will be applicable to advanced pressurized water reactors, and specifically to the System 80+ Standard Design. The resolution of issues for the System 80+ Standard Design will be substantiated, as required, by plant specific evaluations based on deterministic analysis and PRA. Topic Papers will be reviewed prior to submittal to the NRC by an Industry Technical Advisory Group organized by ARSAP. Figure C-1 shows the severe accident resolution process.

The proposed Topic Papers have been divided by ARSAP into six categories corresponding to subject area and sequence of preparation. The categories and preliminary schedule for preparation of Topic Papers are shown in Figure C-2. Table C-1 provides a preliminary list of the issues that are expected to be included in each category.

Combustion Engineering and ARSAP have chosen the Modular Accident Analysis Program (MAAP) Version 3B as the methodology for deterministic analysis of the System 80+ Standard Design to support resolution of severe accident issues [severe accidents that are found to occur at a frequency below an established cut-off frequency (e.g., 1×10^{-8} per reactor year) will not be analyzed deterministically]. This methodology will be applied for design-specific analyses of accident initiation, progression, and containment response. MAAP 3B is a best-estimate method which uses a modular format for modeling plant systems and for predicting a quantified release of radioactive materials from containment corresponding to different postulated accident sequences. It will also be used in sensitivity analyses to investigate the effectiveness of alternative design features for mitigation of degraded core accidents.

It should be emphasize here that NRC approval of the MAAP code is not required. Technical disagreements between the MAAP 3B results and NRC . thods will be addressed on a case-by-case basis in accordance with the review procedures outlined in Section 5.0 of the Licensing Review Bases.

Table C-1
Preliminary Listing of ARSAP Topic Papers

Set 1 RESOLVED IDCOR/NRC ISSUES - APPLICABILITY TO ALWRS
- o Reactor coolant system natural circulation (IDCOR Issue 2)
- o In-vessel steam explosions and alpha mode failure (IDCOR Issue 7)
- o Ex-vessel heat transfer models from molten core to concrete (IDCOR Issue 10)
- o Fission product release prior to vessel failure (IDCOR Issue 1)
- o Release model for control rod materials (IDCOR Issue 3)
- o Fission product and aerosol deposition from primary system (IDCOR Issue 3)
- o Ex-vessel fission product release (during core-concrete interactions) (IDCOR Issue 9)
- o Fission product and aerosol deposition in containment (IDCOR Issue 12)
- o Amount and time of suppression pool bypass (IDCOR Issue 13a)
- o Revaporization of fission products (IDCOR 11)
- o Secondary containment performance (IDCOR Issue 16) (Resolved by design)
- o Modeling of emergency response (IDCOR Issue 14)

Set 2 PLANT RESPONSE UNDER SEVERE ACCIDENT CONDITIONS
- o In-vessel hydrogen generation (IDCOR Issue 5)
- o Core melt progression and vessel failure (IDCOR Issue 6)
- o Direct containment heating by ejected core materials (IDCOR Issue 8)
- o Containment performance (capability, failure modes, isolation, bypass) (IDCOR Issue 15)
- o Hydrogen ignition and burning (IDCOR Issue 17)
- o Fission product release during high pressure core ejection

Set 3 PROBABILISTIC METHODS
- o External events -- seismic (Fire and flood resolved by design)
- o Human factors -- required operator actions
- o Human factors -- unexpected operator actions with potential adverse effect
- o Human factors -- quantification of human error probabilities
- o Success criteria -- partial success and mission time
- o Common cause failures
- o Identification of dominant sequences

Set 4 RISK REDUCTION MEASURES
- o Essential equipment performance (IDCOR Issue 18)
- o Severe accident management -- plant equipment/information system capability
- o Severe accident management -- conditions for safe stable states
- o Mitigation features

Set 5 RISK RESULTS
- o Consensus on integrated severe accident analysis code capability, validation, and application
- o Safety goal implementation -- interpretation of goals and usage of PRA results in comparison with goals, including interpretation of uncertainties
- o Uncertainties in plant risk -- effects of system analysis uncertainties
- o Uncertainties in plant risk -- effects of uncertainties in severe accident analysis (Phenomenology, plant damage states, methodology)
- o Uncertainties in plant risk -- treatment of propagation of uncertainties
- o Uncertainties in plant risk -- completeness of choice of sequences and cutoff probabilities

Set 6 APPLICATIONS OF METHODS
- o Effect of severe accident issues on regulations -- probabilistic accident design bases
- o Effect of severe accident issues on regulations -- assessment of regulatory compliance alternatives
- o Effect of severe accident issues on regulations -- effectiveness of technical specifications

FIGURE C-1

THE SEVERE ACCIDENT RESOLUTION PROCESS

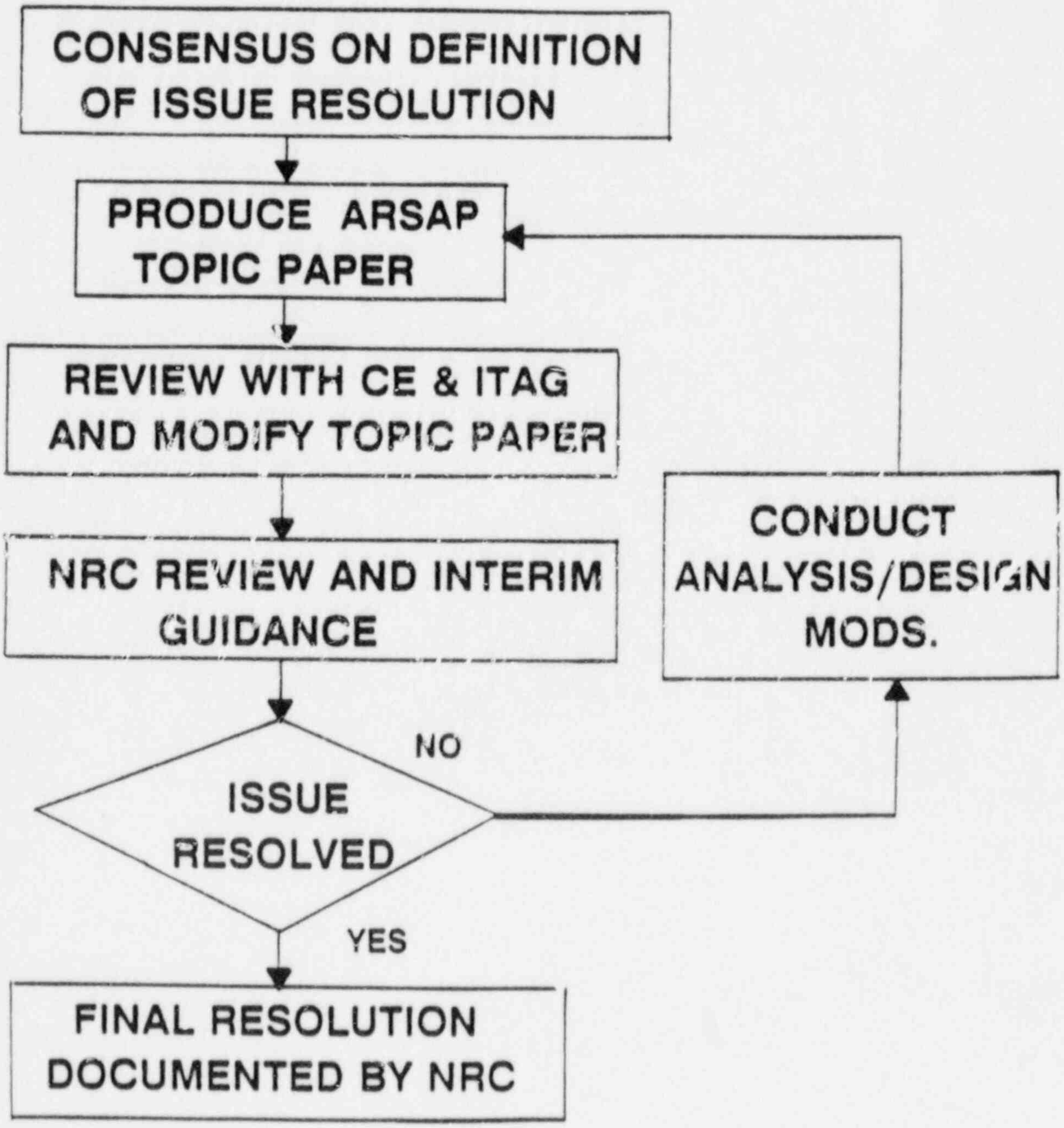
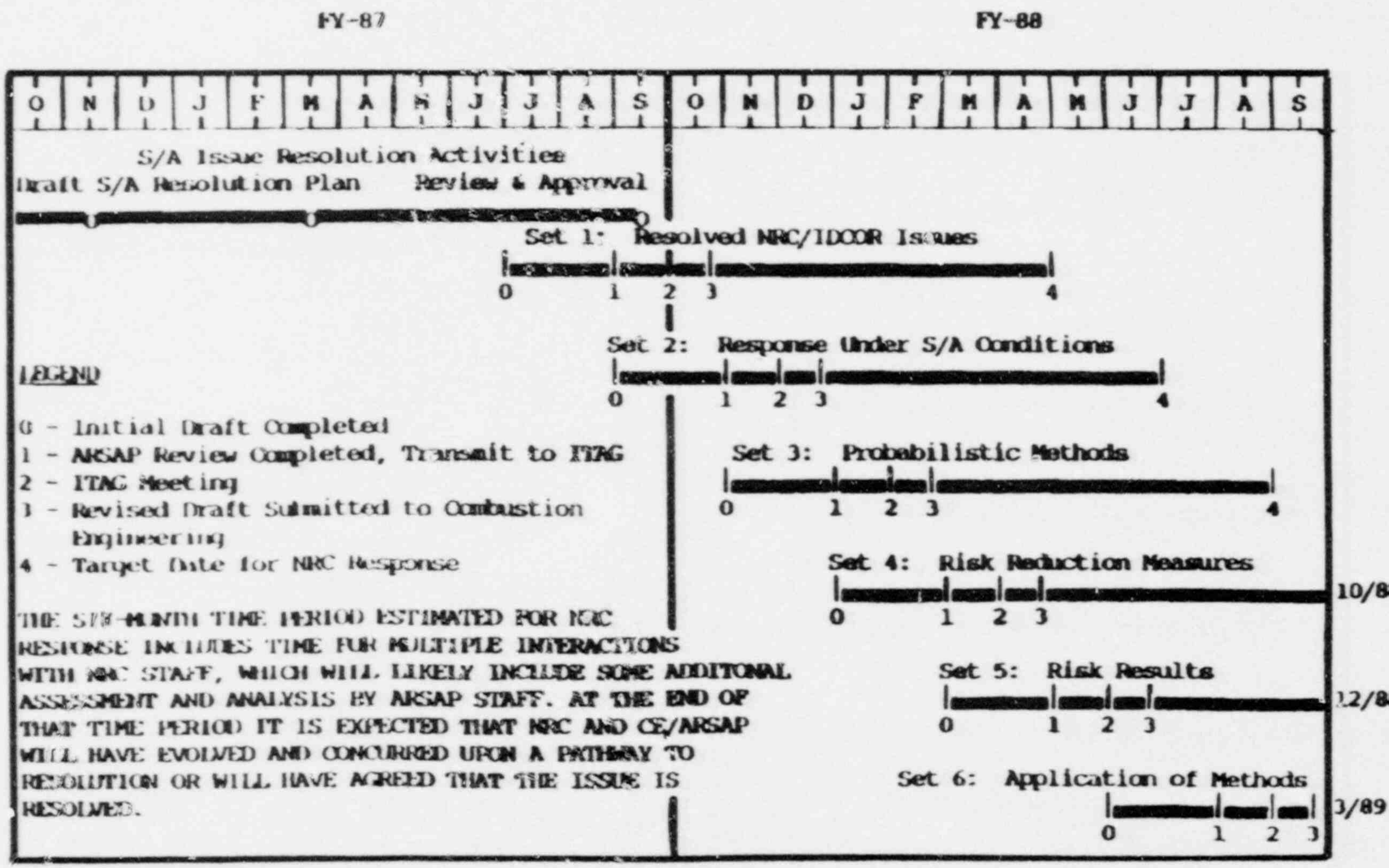


Figure C-2

Identification and Resolution of Severe Accident Issues (Preliminary)



III. Criteria for Degraded Core Evaluation

The resolution of severe accident issues to be documented in Topic Papers will be consistent with NRC guidance on implementation of the SAPS and with the NRC Safety Goal Policy Statement (SGPS). The SGPS includes the general performance guideline that the overall mean frequency of large releases of radioactive material to the environment as a result of reactor accidents should be less than 10^{-6} per year of reactor operation. Procedural criteria for degraded core evaluations are expected to be issued in future regulatory documentation. The following criteria are currently proposed by the NRC staff:

- the evaluation should use realistic prediction of radioactive material releases commensurate with the event;

- for each design, the more likely of severe accidents needs to be considered in the design and licensing of the plant;

- evaluation of severe accident consequences does not need to use conservative engineering practice common for design basis events;

- consequences of more likely severe accidents should not represent a threat to the public; and,

- extremely unlikely events need not be considered in computing consequences, but should be assured of extremely low probability of occurrence.

IV. NRC Review Process

The proposed resolution of severe accident issues for the System 80+ Standard Design will be documented in Topic Papers and submitted for NRC review as an appendix to CESSAR-DC, using the same process as described in Appendix A of this paper for NRC review of Unresolved Safety Issues and Generic Issues. The NRC Staff will provide interim guidance as to the appropriateness of each resolution submitted so that the design process can proceed on schedule. It is possible that the NRC Staff may desire additional information, including results of deterministic analyses for degraded core accidents, to support their review. This information will, therefore, be provided through informal interactions as required. Revision of the Topic Paper submittals will be made as necessary and sufficient information will be provided by Combustion Engineering and ARSAP to enable the resolution of all severe accident issues applicable to the System 80+ Standard Design.

NRC review results will be documented in draft Safety Evaluation Reports (SERs) following completion of initial review resulting in resolution of the issue or agreement on an achievable pathway for resolution. The SERs will address the acceptability of resolutions for severe accident issues including criteria applied for the System 80+ Standard Design and methods of evaluation. The SERs will be finalized upon completion of an integrated review of CESSAR-DC by the NRC staff.

IV. Summary

The System 80+ Standard Design degraded core evaluation will address severe accident issues applicable to advanced pressurized water reactors. The resolution of severe accident issues will be based on the requirement to demonstrate safety acceptability in compliance with the NRC severe accident and safety goal policy statements. Combustion Engineering and ARSAP will propose criteria for resolution of severe accident issues by means of Topic Papers and an appendix to CESSAR-DC submitted on the CESSAR-DC docket. The NRC Staff will provide interim guidance on the appropriateness of the proposed resolution and will request additional information, as required, sufficient for resolution of each issue. Results of NRC review will be documented in the CESSAR-DC Safety Evaluation Report.

APPENDIX D

COMBUSTION ENGINEERING DESIGN CERTIFICATION PROGRAM

INSTRUMENTATION AND CONTROLS

(LATER)

RESPONSE TO NRC COMMENTS ON LRB (OCTOBER 29, 1987 VERSION)

The following is a listing of NRC comments on the October 29, 1987, version of the Licensing Review Bases with reference made to the appropriate section or sections of the current version where Combustion Engineering has addressed that particular comment.

1. a. If there would be technical disagreement between the NRC method of analysis and the MAAP code, C-E should propose an alternate solution.

   Reference: Section 5.0, Appendix B and Appendix C

   b. C-E should propose a method to compute potential consequences of fission product release.

   Reference: Section 7.3.3, Appendix B and Appendix C

2. Severe Accident Goals

   a. C-E should propose a core damage frequency goal.

   Reference: Section 7.3.1

   b. No mitigation of Core Damage is proposed. C-E should address:

      1. Measures to reduce early failure of containment
      2. Measures to accommodate hydrogen production
      3. Heat removal systems for containment
      4. Measures to prevent hydrogen detonation

   Reference: Section 7.3.2

   c. C-E should address dose limits and maximum probability per year of experiencing the limits considering internal and external events. Containment design should have a failure frequency of equal to or less than 1/10.

   Reference: Section 7.3.3

3. C-E should address Physical Security. Consideration should be given to specific design requirements such as:

   Physical Security Organization
   Detection Aids

Testing and Maintenance
Communication Requirements
Response Requirements

Reference: Section 8.1

4. C-E should provide discussions on site parameters or soil-structure interaction analysis.

   Reference: Section 8.2

5. C-E should address details on defining major design components and include the result of sufficient engineering to identify:

   Design basis criteria
   Analysis and design methods
   Physical arrangement of auxiliary, BOP and NSSS systems
   Physical arrangement of plant
   Performance specifications

   Reference: Section 8.3

6. C-E should address details on instrumentation and controls.

   Reference: Section 8.6 and Appendix D (later)

7. C-E should address details on designing for maintenance and surveillance.

   Reference: Section 8.8

8. C-E should address QA.

   Reference: Section 8.4

9. C-E's Safety Goal Policy Statement provides no concrete commitment. C-E should be more specific.

   Reference: Section 8.9

10. C-E should address the application of 10 CFR 50.34(g), the Standard Review Plan, in the review.

    Reference: Section 3.2 and Section 5.1

11. The LRB should define the scope of the System 80+ Design which is proposed for design certification, i.e., those systems which will be included and those systems which represent the remainder of the plant.

    Reference: Section 1.0

12. The LRB should discuss in greater detail the Standard Functional Requirements of the balance of the plant.

    Reference: Section 8.5