Enclosure
LD-88-055

COMBUSTION ENGINEERING, INC.

SYSTEM 80+™ STANDARD DESIGN

DESIGN CERTIFICATION
LICENSING REVIEW BASES

July 15, 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.

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 System 80+ Standard Design will, therefore, include all buildings, structures, systems and components that can significantly affect the safe operation of the plant. Accordingly, the Staff's review of CESSAR-DC will close out all questions concerning the System 80+ Standard Design and will establish the acceptance criteria for site-specific construction verification.

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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 referred to as the Nuclear Power Module) as well as detailed Standardized Functional Descriptions for all other buildings, structures, systems and components 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.

Both Combustion Engineering and the NRC Staff believe 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).

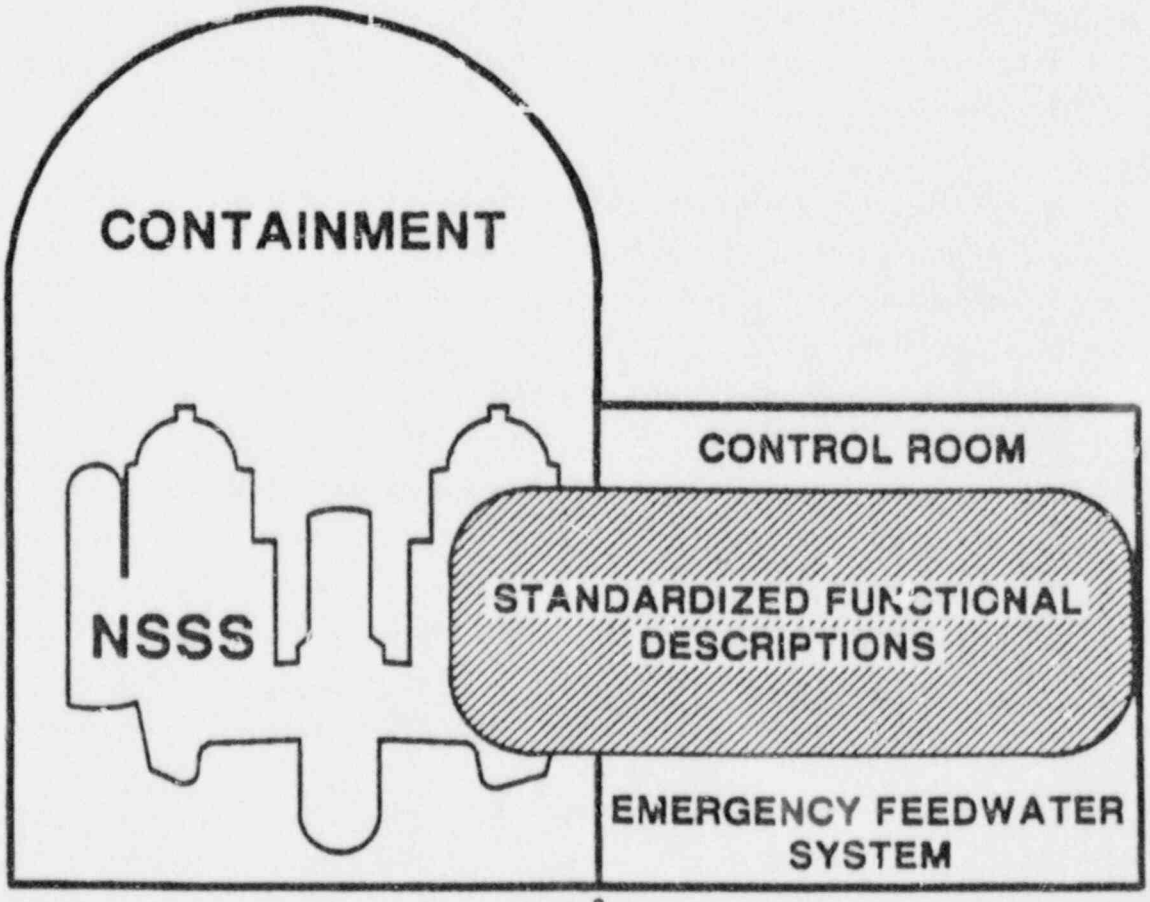
FIGURE 1
SCOPE OF THE SYSTEM 80+ STANDARD DESIGN

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 Protective 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 buildings, structures, systems and components to enable the Staff to reach the required public health and safety determination for the System 80+ Standard Design.



1.1 Scope and Content of CESSAR-DC

The System 80+™ Standard Design will use, as a starting point, the System 80® design covered by the current FDA and described in CESSAR-F. In accordance with the guidance provided in the Commission's Severe Accident Policy for standard designs already holding an FDA, it is unnecessary for the NRC Staff to repeat its review of the design against existing regulations. In cases where the design has been modified, however, it will be appropriate for the Staff to assure that its previous decisions are still applicable or modify them accordingly.

The System 80+ Standard Design, as shown in Figure 1, includes the Nuclear Power Module and Standardized Functional Descriptions. The expansion of the System 80 design to include the Nuclear Power Module and detailed Standardized Functional Descriptions will ensure that a sufficient level of information pertaining to all buildings, structures, systems, and components that can significantly affect the safe operation of the plant will be provided. This will ensure that all safety issues for the System 80+ Standard Design are fully addressed and that all regulatory requirements are accounted for during the Design Certification process. The Staff's review of CESSAR-DC, therefore, will close out all questions concerning the System 80+ Standard Design and will address the tests, analyses and inspections that are necessary to provide reasonable assurance that the plant can be built and operated within the specifications of the certified design.

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.

Since Combustion Engineering wishes to obtain an FDA Amendment and a Design Certification for the System 80+ Standard Design before any applicant, site or equipment suppliers are identified, Combustion Engineering will provide the necessary level of detailed information to enable the Staff to complete its review without preempting competitive bidding on any future project that references the certified design. The corresponding format and contents of CESSAR-DC are described in Section 3.0 and 8.3.

According to Section B.3.b(1) of the Commission's Severe Accident Policy, the requirements of 10 CFR 50.34(g) are not applicable to CESSAR-DC, since the FDA for the System 80+ Standard Design will be an amendment to the System 80 FDA for the purpose of complying with the Severe Accident Policy. Where the System 80 design has been modified, however, Combustion Engineering will identify to the Staff any non-conformances with the SRP.

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 reference design need consider only (1) whether the site envelope parameters of the certified design fall within the requirements of the specified construction site, (2) the applicant's proposed means of assuring that plant construction will conform to the certified design requirements, and (3) a final determination (based on compliance reviews/audits during construction) that the plant has been constructed and can be operated in compliance with the design details and acceptance criteria certified by the Commission. No further review of the referenced design will be required.

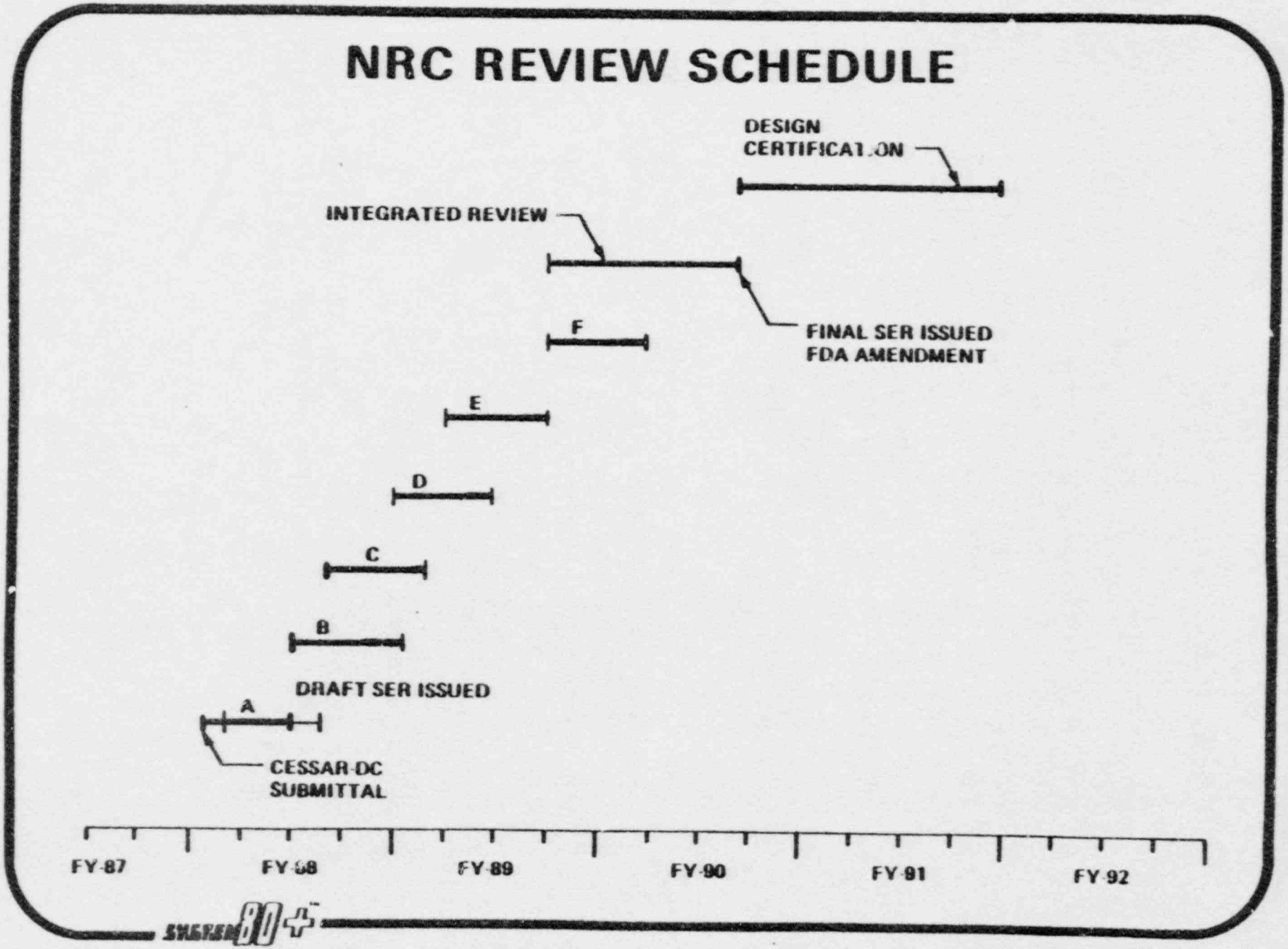
2.0 SCHEDULE

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 a goal of an average review period of six months per submittal group. This average review period is a goal which should be met whenever practical in order to meet the objectives of FDA Amendment by June 1990 and Design Certification by September 1991.

Table 1
CESSAR-DC Submittal Schedule

Submittal Group	Description	Revision of CESSAR-F Chapter (Sections)	Implementation of EPRI Chapter	CESSAR-DC Submittal Date	Draft SER Issued
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 Core, Reactor Coolant System, Chemical and Volume Control System, Process Sampling System	4 (all) 5 (5.1,5.2,5.4) 9 (9.3)	3 & 4	Submitted April 1988	Oct. 1988
C	Shutdown Cooling System, Safety Injection System, Emergency Feedwater System	5 (5.1,5.2,5.3,5.4) 6 (6.1,6.3,6.6) 10 (10.4)	5	Submitted June 1988	Dec. 1988
D	Building Design & Site Arrangements, Safety Depress. System, Instrumentation & Control Systems, Human Factors Engineering	2 (all), 3 (3.1-3.5, 3.10,3.11,3A) 5 (5.2,5.3,5A,5B,5C) 6 (6.4,6.5,6.7,6A,6B) 7 (7.1,7.4,7.5,7.6,7.7) 10 (10.3), 13 (18.1, 18.2,18.3)	6 & 10	Sept. 1988	March 1989
E	Leak-Before-Break & Press. Thermal Shock Eval., Reactor Protective System, Elec. Power, Fuel Handling Systems, Rad. Waste System, Nuplex 80+ Evaluation	1 (all), 3 (3.6,3.7,3.8, 3.9), 5 (5D), 7 (7.2, 7.3), 8 (all) 9 (9.1,9.2,9.4,9.5) 10 (10.2,10.4),11-14 (all) 17 (all), 18 (18.4)	7 - 13 (except 10)	Dec. 1988	June 1989
F	Safety Analyses, USI-GI Resolution, Tech. Specs., PRA, Degraded Core Performance	6 (6.2,6.3) 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
DESIGN CERTIFICATION
INTEGRATED REVIEW
FINAL SER ISSUED
FDA AMENDMENT
F
E
D
C
B
DRAFT SER ISSUED
A
CESSAR-DC SUBMITTAL
FY-87
FY-88
FY-89
FY-90
FY-91
FY-92
SYSTEM 80+

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

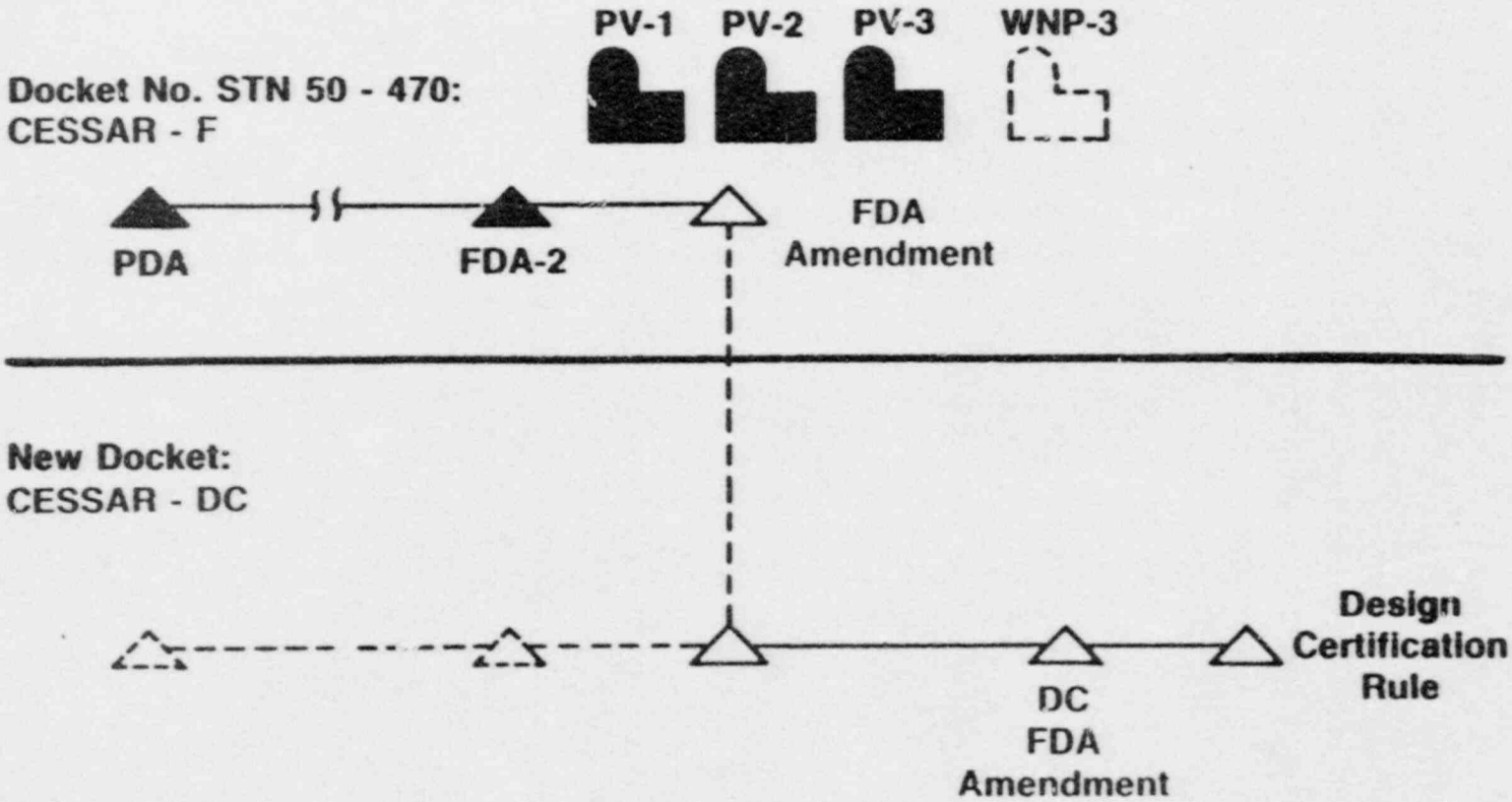
The safety review of the System 80+ Standard Design for Design Certification will be performed by NRC reviewers who are accustomed to working with the format and organization of the NRC's Standard Review Plan (NUREG 0800) and Regulatory Guide 1.70 (Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants). 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.

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.

FIGURE 3

DUAL DOCKET APPROACH



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, 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 NRC STAFF REVIEW

5.1 Overview

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. 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 8.5)] 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 Procedure

The staff will follow its review procedures in the SRP, supplemented and modified as follows:

(1) CESSAR-DC is to be submitted in groups as shown in Table 1. Correspondingly, the staff SER will also be issued in draft form, in sections in accordance with the schedule also shown in Table 1. The draft SER sections will be made publicly available.

(2) At the completion of the review of the individual SAR chapters, the staff will perform an integrated review of the application. This review will complement the Probabilistic Risk Assessment (PRA) review, 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.

(3) It will be important to carefully document open or unresolved issues that may be identified early in the review process, but which cannot be resolved until the completion of later chapters. Each draft SER section will contain a description of such issues. In addition, with the submittal of each chapter of CESSAR-DC, Combustion Engineering is to provide an updated checklist which identifies outstanding issues and the future chapter(s) in which resolution is anticipated.

(4) Each draft SER will contain a target schedule for closing outstanding SER issues that is compatible with the target date for the FDA Amendment.

6.0 ACRS PARTICIPATION

One step in the design review of a standard plant is the independent review by the Advisory Committee on Reactor Safeguards (ACRS). Periodic reviews will address the safety aspects of the design changes and/or design enhancements on matters selected by the 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 provide defense in depth by striking 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

Combustion Engineering will comply with all applicable Commission regulations, including those listed in 10 CFR 50.34(f), applicable to the System 80+ Standard Design. In special cases [e.g. 10 CFR 50.34(f)(2)(i)], this compliance may take the form of explicitly placing a requirement to comply with a particular regulation on any future applicant that references the System 80+ Standard Design. As discussed in Section 1.1, however, the requirements of 10 CFR 50.34(g) do not apply to CESSAR-DC.

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 GIs

The process for developing the resolution of USIs and GIs 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 for severe accident performance criteria. These goals are consistent with the guidance of the NRC's Severe Accident and Safety Goal Policies.

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 goals stated in the following sub-sections were developed to meet those utility requirements while remaining consistent with NRC guidance. Combustion Engineering will demonstrate that the System 80+ Standard Design meets the following design goals by submitting a Level III PRA as discussed in Appendix B.

7.3.1 Prevention of Core Damage

For the System 80+ PRA, Combustion Engineering has adopted the following criteria for potential severe core damage.

A potential for severe core damage shall be assumed to exist if and only if both of the following have occurred:

(A) The collapsed level in the RCS has decreased such that active fuel in the core has been uncovered; and,

(B) A temperature of 2200°F or higher is reached in any node of the core as defined in a realistic thermal-hydraulic calculation.

If the above criteria for potential severe core damage are exceeded, predictions of actual core damage and resulting radioactive releases will be calculated using the MAAP code. The above criteria are consistent with the EPRI definition provided in Section 1.2 of the EPRI ALWR Requirements Document. It is Combustion Engineering's goal that the estimated mean annual core damage frequency (including both internal and external events) will be less than 1×10^{-5} events per reactor year.

Any breach of the containment during or following a core damage event is considered to be a failure of the containment. It is Combustion Engineering's goal that no containment failure modes shall exist that lead to offsite doses in excess of the design goal (Section 7.3.3), with a mean frequency of greater than once per million reactor years.

With regard to meteorology, the methods and assumptions employed in the analysis of environmental transport consequences (plume size/wind direction/wind speed/wind shift probability/adverse or expected weather), population distribution (probability of individual seeing plume/location of individual(s) during release), and time of exposure will be consistent with the guidance found in NUREG/CR-2300, dated January, 1983, and NUREG/CR-2815, dated August, 1985.

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

Accordingly, Combustion Engineering has adopted the following design goal for the System 80+ Standard Design.

In the event of a severe accident, the dose beyond a one-half mile radius from the reactor shall not exceed 25 rem. The mean frequency of

occurrence for higher offsite doses shall be less than once per million reactor years, considering both internal and external events.

An industry effort, sponsored by EPRI, has evaluated the guidance of the Safety Goal and Severe Accident Policy Statements and documented a quantitative design goal for addressing the portion of these policies dealing with large radioactive releases resulting from a severe core accident (Chapter 1 of the EPRI ALWR Requirements Document). The Combustion Engineering design goal is consistent with the EPRI design goal.

Probabilistic Risk Assessment (PRA), using mean values, will be used by Combustion Engineering to demonstrate that the System 80+ Standard Design achieves these design goals. The System 80+ Level III PRA will be performed by modifying and extending the baseline System 80^R PRA. The accident sequences to be quantitatively evaluated will be of the type and number listed in Tables 7.2-1 to 7.2-9 of the baseline PRA report [Enclosure to Letter, LD-88-008, A. E. Scherer (C-E) to G. S. Vissing (NRC), dated January 22, 1988]. That report also provides detailed descriptions of the system modeling methods, analysis ground rules, and computer codes that were used (Section 2.0). The PRA evaluation process for the System 80+ Standard Design will be similar to that described in the baseline PRA report.

External events will be considered in the System 80+ PRA. There is currently an on-going Advanced Reactor Severe Accident Program (ARSAP) task to identify the degree to which each external event category should be quantitatively evaluated in the System 80+ PRA. This task should be completed by the end of 1988. Combustion Engineering will then be in a position to adopt the ARSAP results.

Sabotage will be considered in the design by identifying those design features which minimize the potential for sabotage. In particular, Combustion Engineering will utilize physical separation of safety trains as well as existing nuclear security design practices to minimize the

risk of sabotage. Combustion Engineering will also address appropriate NRC guidance, either through design features or through requirements in the Standardized Functional Descriptions. Sabotage will not be addressed quantitatively in the System 80+ PRA.

In summary, the use of PRA, in conjunction with industry and NRC guidance, will determine whether the Combustion Engineering design goals for severe accidents have been achieved.

8.0 ADDITIONAL ISSUES

8.1 Physical Security and Sabotage

The Severe Accident Policy states that "...sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized as special considerations in the design and in the operating procedures developed for new plants." This statement will be addressed in Combustion Engineering's Sabotage Protection Program. The basic elements of this program will be a literature review, sabotage analysis, design assessment, and frequent feedback from NRC Staff.

In addition, 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. The basis for this guidance will be as defined in 10 CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage," and other applicable portions of 10 CFR 73. 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.

CESSAR-DC will include enough information, through either design details or detailed Standardized Functional Descriptions, to ensure the existence of adequate physical barriers to protect vital equipment in accordance with 10 CFR 73.55(c), "Physical Barriers," and to identify access control points to all vital areas in accordance with 10 CFR 73.55(d), "Access Requirements."

CESSAR-DC will identify, through detailed Standardized Functional Descriptions, the Physical Security System design requirements that must

be satisfied by the applicant. The site-specific application will then address the following sections of 10 CFR 73.55:

(b) Physical Security Organization
(e) Detection Aids
(f) Communication Requirements
(g) Testing and Maintenance
(h) Response Requirements

The design requirements are to include reference to existing NRC documents such as Regulatory Guide 5.44, "Perimeter Intrusion Alarm Systems" and NUREG-0908, "Acceptance Criteria Evaluation of Nuclear Power Reactor Security Plans", as well as to industry standards such as IEEE-692-1986, "IEEE Standard Criteria for Security Systems for Nuclear Power Generating Stations."

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.

8.3 Completeness of Design Documentation

CESSAR-DC is to provide essentially complete design information. The term "essentially complete" is defined as follows:

(1) CESSAR-DC will define the major design components and include the results of sufficient engineering to identify, as appropriate:

a. design basis criteria
b. analysis and design methods
c. functional design and physical arrangement of systems

d. physical arrangements sufficient to accommodate systems and components
e. functional and/or performance specifications
f. acceptance/test requirements
g. risk assessment methodology

(2) Design documentation of systems, structures, and components should include, as appropriate:

a. design basis criteria
b. plant general arrangements
c. process and instrumentation diagrams
d. control logic diagrams
e. system functional descriptions and supporting studies and analyses
f. sufficient detail to permit preparation of component specifications, including acceptance criteria and test requirements
g. sufficient detail to permit preparation of construction/installation specifications, including acceptance criteria and test requirements
h. program for the assurance of quality
i. design-related aspects for the emergency plans
j. supporting design data
k. design-related aspects of the physical security program
l. ALARA/radiation protection plan
m. accident analyses
n. technical specifications
o. 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. Combustion Engineering will also define those related tests, inspections and acceptance criteria that are necessary to assure that the designs are properly implemented in the plant. These tests,

inspections and acceptance criteria are intended to be implemented and verified in a series of reviews by the applicant during construction and pre-operation. The staff will monitor the performance of these reviews and implementation of the design through its inspection program.

The degree of detail necessary for providing an essentially complete design is to be that which is suitable for obtaining specific equipment or construction bids and to demonstrate conformance to the safety limits and criteria.

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. Combustion Engineering intends to submit an updated revision to its previous topical report. This revision will be reviewed to verify that it has not diminished Combustion Engineering's commitment to quality and that it is still in compliance with the provisions of Appendix B of 10 CFR 50.

8.5 Standardized Functional Descriptions

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 SFD 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 future applications referencing the System 80+ Standard Design can be limited to a compliance review as discussed in Section 1.2.

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 I&E Bulletins

All Generic Letters and I&E Bulletins are being reviewed as part of the EPRI ALWR Requirements program. Combustion Engineering will utilize EPRI's review of these items. It is anticipated that all Generic Letters and I&E Bulletins applicable to the System 80+ Standard Design will be addressed through implementation of EPRI ALWR Design Requirements and/or through resolution of all applicable USIs and GIs. A separate review of Generic Letters and I&E Bulletins, therefore, will not be necessary.

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 and 21, 1986, the Commission published a Policy Statement on "Safety Goals for the Operation of Nuclear Power Plants" (51 FR 28044 and 51 FR 30028). 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.

Combustion Engineering will comply with those implementation requirements that are developed by the NRC which are applicable to the System 80+ Standard Design. Combustion Engineering will apply the severe accident performance goals of Section 7.3 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.

9.0 FINAL DESIGN APPROVAL

In August 1985, Combustion Engineering requested that the current CESSAR-F FDA (FDA-2) be amended to permit forward referenceability to those plants which hold a preliminary design approval in accordance with the NRC Severe Accident Policy Statement (the Severe Accident Policy requirements would be resolved on a plant specific basis for each applicant for a full power license). Upon completion of NRC Staff review of that request, the Staff will issue a forward referenceable FDA Amendment (FDA-2, Amendment 1) that will be applicable to Docket No. 50-470, 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 permit forward referenceability in complete compliance with the Severe Accident Policy Statement (the Severe Accident Policy requirements would be resolved for the System 80+ Standard Design). The amended FDA (FDA-2, Amendment 2) will be the basis for a System 80+ Design Certification Rule and will be applicable to the new docket only.

An FDA means that the design is acceptable for incorporation by reference in individual applications for construction permits and/or operating licenses. The staff and the ACRS intend to use and rely on the approved final design in their reviews of referencing applications. However, an approved final design is still subject to litigation in individual licensing proceedings on referencing applications. An FDA is a prerequisite for a design certification.

10.0 DESIGN CERTIFICATION

10.1 Introduction

The Commission revised its 1978 policy statement on the standardization of nuclear power plant designs on September 15, 1988 (52 FR 34884). The Commission also is developing proposed regulations that will address licensing reform and standardization and provide a regulatory framework for implementation of the standardization policy, including Commission certification of standard designs by rulemaking. Since design certification is the ultimate goal of the System 80+ Design Certification program, and since the focus of the revised policy statement and proposed regulations is reference system design certification, the essence of these proposals, and Combustion Engineering's commitment to them, is summarized here. It should be noted, however, that the Commission has not yet acted on proposed regulations and that they are subject to change.

The Commission's revised policy statement encourages the use of standard plant designs in all future license applications. The Commission believes that the use of certified standard plant designs can benefit public health and safety by:

(1) Concentrating resources on specific design approaches without stifling ingenuity;

(2) Stimulating standardized programs of construction practice, quality assurance and personnel training; and,

(3) Fostering more effective maintenance and improved operation.

The staff believes that the use of such standardized designs can also permit more effective and efficient licensing and inspection by the NRC.

10.2 Design Certification Concept

The design certification concept, as described in the Commission's revised standardization policy statement, 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 the staff issues an FDA (an Amendment to FDA-2 for the System 80+ Standard Design) and a rulemaking proceeding is completed. The design certification means that the portions of the nuclear power plant design that have been reviewed are acceptable for incorporation by reference in an individual license application. The conclusions of the certification rulemaking would be used and relied on by the staff, the ACRS, the hearing boards, and the Commission in their reviews of applications that reference the design. The certified design would not be subject to litigation in individual licensing proceedings, except as provided in 10 CFR 2.758.

Under the staff proposed regulations implementing the Standardization Policy, the Commission could certify the System 80+ Standard Design for referencing by applicants for a period of 10 years. Renewal of the design certification could be granted for an additional period of up to ten years unless the Commission found that the design would not comply with the Commission's regulations. Applicants could reference the certified design in applications for CPs and OLs docketed during the period beginning with the docketing date of the CESSAR-DC revisions and ending at the expiration date of the design certification. However, no CP or OL could be issued for an application referencing the System 80+ Standard Design until the FDA Amendment is issued as described in Section 9.0.

10.3 Completeness of Scope and Design Detail

The System 80+ Standard Design submitted for design certification will include the Nuclear Power Module (the major portion of a complete nuclear power plant design) and Standardized Functional Descriptions. The System

80+ Standard Design is also to demonstrate compliance with the licensing criteria for new plant designs set forth in the Commission's Severe Accident Policy Statement (Section 7). CESSAR-DC is to address the tests, analyses, and inspections that are necessary to provide reasonable assurance that the plant can be built and operated within the specifications of the certified design. For those individual aspects of the design where safety-related structures and components differ from those of existing designs, empirical information is to be included as part of the application for design certification.

CESSAR-DC is to include information that will permit construction verification and compliance. This will permit reviews during the construction and startup phases of the plant and will eliminate the need for further design reviews except to verify that the Standardized Functional Descriptions have been satisfied.

10.4 Changes to Approved and Certified Designs

Both Combustion Engineering and the NRC Staff believe that standardization will be best achieved if changes to approved or certified designs are kept to a minimum. Nevertheless, there are situations in which changes may be needed or desirable. It is the staff's intent that after issuance of the design certification, the Commission would require backfitting only when it determines, using the standards in 10 CFR 50.109 and the results of the Design Certification rulemaking, that a substantial cost beneficial increase in the overall protection of the public health and safety would result.

Combustion Engineering may request modifications to the certified design by applying for an amendment to the design certification in accordance with the proposed regulations.

10.5 Review Fees

The Commission has the ability under existing law to enhance the standardization option by adjusting the fees it charges for the review of standard designs. Therefore, in addition to not charging an application fee, the NRC will defer any fees associated with review of the application, pending the filing of applications for construction permits or combined licenses referencing the certified standard design. Under the Staff's proposed regulations, any outstanding fees will become due and payable by the holder of the design certification at the end of the certification period. It is further suggested that fees for the renewal of a standard design certification be assessed in the same manner.

Although general agreement and support has been expressed for the concept of fee deferral, Combustion Engineering has expressed concern over the existence of an unlimited fee category and the obstacle it presents to the pursuit of a pre-approved standard design since it would essentially allow the Staff to accumulate costs without limit during the review of the System 80+ Standard Design. The resulting unspecified deferred liability would make it impossible for Combustion Engineering to accurately estimate the cost of the design certification effort and would add an undesirable administrative hurdle to the filing of its formal application.

Combustion Engineering, therefore, has requested that the Commission establish a specific fixed fee category for reviews and hearings related to Design Certifications. With respect to the magnitude of the fixed fee, Combustion Engineering has further proposed that it be waived for plant designs which already hold an FDA (such as the System 80 design). This is due to their position that the design has already been reviewed, approved, and an FDA issued and thus the NRC has determined that the design meets all of the Commission's applicable rules and regulations. All that is left, therefore, is to conduct a review to determine if the design will comply with the additional requirements of the Standardization and Severe Accident Policy Statements. That effort should demand significantly less time and resources than the original

System 80 FDA review for which the fee has already been assessed. If not waived entirely, Combustion Engineering has suggested that the review fee, at most, be fixed at a small fraction of its FDA fee.

The Commission has stated its intention to address this matter in the ongoing standardization rulemaking process. Combustion Engineering has, therefore, requested that a means be found by which they could reserve their rights in this matter while submitting material for the Staff's review of the System 80+ Standard Design. This will make it possible to resolve the fee issue in parallel with Combustion Engineering's formal submittals and thus allow the technical review of the design to proceed unencumbered. The Staff has this request under review.

10.6 Rulemaking

Appendix O to 10 CFR 50 provides the opportunity for the Commission to approve the System 80+ Standard Design in a rulemaking proceeding. The regulations that are currently under development will specify the procedures to be used for the rulemaking. In general, however, it is anticipated that these regulations will require that, upon receipt of a request from Combustion Engineering, a notice would be published in the Federal Register announcing the request for a Design Certification for the System 80+ Standard Design. The notice would set out the matters at issue, as specifically as possible, and the hearing procedures to be used if the Commission decided to hold a hearing. The notice would further request that all persons wishing to participate in a hearing notify the Commission within a stated period of time. As a condition to participating in a hearing, however, intervenors could be required to state the issues they wish to have considered at the hearing and to commit to providing expert testimony on those issues. Written comments could be invited from those not intending to participate in the hearings. As a result of responses to the notice, or on its own initiative, the Commission could then hold hearings on the proposed rulemaking.

The views of the ACRS would be sought and considered. The ACRS would review the design before the rulemaking, and the results of the ACRS review would be made available when the proposed rule is announced.

After any hearing, the Commission would review the record of the rulemaking, including both the hearing record and any other written comments. The notice of final rulemaking would include responses to written comments and the resolution of issues considered at a hearing.

10.7 Renewal of Certifications

Under the staff's proposal, the Commission could certify the System 80+ Standard Design for referencing by applicants for a period of 10 years. Additionally, before the expiration of the design certification, Combustion Engineering could apply for certification renewal. The design certification could be renewed for an additional 10-year period, provided the design complies with the Commission's regulations in effect at the time of the renewal application.

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 anticipated that Combustion Engineering will provide the NRC Staff with the information necessary to close out all applicable review issues so that a Design Certification rulemaking can be concluded without open issues or conditions.

II. Acceptance Criteria for Resolution of USIs and GIs

The USIs and GIs that are 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. Others have been prioritized into categories of USI, and High-, Medium-Priority, and Nearly Resolved Generic Issues. Based on the GIMCS listings, the Combustion Engineering Design Certification Program will identify and resolve the USIs and the High- and Medium-Priority GIs which are 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 evaluate input from various sources (described below) and each applicable safety issue will be resolved and documented on the CESSAR-DC docket. Some issues have already been resolved by the NRC and -in these cases- Combustion Engineering will implement, to the maximum extent possible, the NRC's proposed 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+, Combustion Engineering 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 comments. The Combustion Engineering 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 based on documented criteria which have been reviewed by the 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, Combustion Engineering 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, Combustion Engineering will develop acceptance criteria and resolutions specific to the System 80+ Standard Design and will obtain NRC approval through documentation in CESSAR-DC.

III. NRC Review Process and Documentation

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 Bases 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. Combustion Engineering 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.	07.4
003	SETPOINT DRIFT IN INSTRUMENTATION	GSI	NEARLY RES	07.1.2
012	BWR JET PUMP INTEGRITY	GSI	MEDIUM	NA
014	PWR PIPE CRACKS	GSI	NEARLY RES	05.2.3
020	EFFECTS OF ELECTROMAGNETIC PULSE ON NUCLEAR PLANT SYSTEMS	GSI	NEARLY RES	07.1
021	VIBRATION QUALIFICATION OF EQUIPMENT	GSI	DROP	03.10
022	INADVERTANT BORON DILUTION EVENTS	GSI	NEARLY RES	05.4.6
023	REACTOR COOLANT PUMP SEAL FAILURES	GSI	HIGH	05.4.1
024	AUTOMATIC EMERGENCY CORE COOLING SYSTEM SWITCH TO RECIRCULATION	GSI	TO BE REP.	06.3
029	BOLTING DEGRADATION OR FAILURES IN NUCLEAR PLANTS	GSI	HIGH	05.[illegible]
036	LOSS OF SERVICE WATER	GSI	NEARLY RES	09.2
038	POTENTIAL RECIRCULATION FAILURE AS A CONSEQUENCE OF CONTAINMENT PAINT OR DEBRIS	GSI	TO BE DET.	06.1
040	SAFETY CONCERNS ASSOC. WITH BREAKS IN THE BWR SCRAM SYSTEM	GSI	NEARLY RES	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
045	INOPERABILITY OF INSTRUMENTS DUE TO EXTREME COLD WEATHER	GSI	NEARLY RES	07.1
048	LCO FOR CLASS 1E VITAL INSTRUMENT BUSES IN OPERATING REACTORS	GSI	NEARLY RES	016
049	INTERLOCKS AND LCO'S FOR REDUNDANT CLASS 1E TIE BREAKER	GSI	MEDIUM	07 OR 08
050	REACTOR VESSEL LEVEL IN BWRS	GSI	NEARLY RES	NA
051	PROP.REQ.FOR IMPROVING REL.OF OPEN CYCLE SER.WTR	GSI	MEDIUM	09
055	FAILURE OF CLASS 1E SAFETY RELATED SWITCHGEAR CIRCUIT BREAKER TO CLOSE ON DEMAND	GSI	TO BE REP.	08.3
057	EFFECTS OF FIRE PROTECTION SYSTEM ACTUATION ON SAFETY RELATED EQUIPMENT	GSI	TO BE DET.	07 OR 09
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.	05.2.
063	USE OF EQUIPMENT NOT CLASSIFIED AS ESSENTIAL TO SAFETY IN BWR TRANSIENT ANALYSIS	GSI	TO BE DET.	NA
065	PROBABILITY OF CORE MELT DUE TO COMPONENT COOLING WATER SYSTEM FAILURES	GSI	HIGH	PRA A1

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
066	STEAM GENERATOR REQUIREMENTS	GSI	NEARLY RES	05.4.2
067	STEAM GENERATOR STAFF ACTIONS	GSI	MEDIUM	05.2
067.7	STEAM GENERATOR STAFF ACTIONS-EDDY CURRENT TESTS	GSI	MEDIUM	05.2.4
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	05
071	FAILURE OF RESIN DEMINERALIZER SYSTEMS AND THEIR EFFECTS ON PLANT SAFETY	GSI	TO BE DET.	09.3
072	CONTROL ROD DRIVE GUIDE TUBE SUPPORT PIN FAILURES	GSI	TO BE DET.	04.[illegible]
073	DETACHED THERMAL SLEEVES	GSI	TO BE DET.	05.4.[illegible]
075	GEN. IMPLICATIONS OF ATWS EVENTS AT SALEM	GSI	NEARLY RES	07.1
076	INSTRUMENTATION AND CONTROL POWER INTERACTIONS	GSI	TO BE DET.	07.
077	FLOODING OF SAFETY EQUIPMENT COMPARTMENTS BY BACKFLOW	GSI	HIGH	06

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
078	MONITORING OF FATIGUE TRANSIENT LIMITS FOR REACTOR COOLANT SYSTEM	GSI	TO BE DET.	05.2
079	UNANALYZED REACTOR VESSEL THERMAL STRESS-COOLDOWN	GSI	MEDIUM	05.3.2
082	BEYOND DESIGN BASES ACCIDENTS IN SPENT FUEL POOLS	GSI	MEDIUM	09.1
083	CONTROL ROOM HABITABILITY	GSI	NEARLY RES	18
084	CE PORVS	GSI	NEARLY RES	05.2
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	06
088	EARTHQUAKE AND EMERGENCY PLANNING	GSI	TO BE DET.	02 & 13
089	STIFF PIPE CLAMPS	GSI	TO BE DET.	05.4.7
091	MAIN CRANKSHAFT FAILURE IN TRANSAMERICA DELAVAL EDG'S	GSI	NEARLY RES	08
093	STEAM BINDING OF AUXILIARY FEEDWATER PUMPS	GSI	HIGH	10.9.3
094	ADDITIONAL LTOP FOR LIGHT WATER REACTORS	GSI	HIGH	05.2.
095	LOSS OF EFFECTIVE VOLUME FOR CONTAINMENT RECIRCULATION	GSI	TO BE DET.	06.2
096	RHR SUCTION VALVE TESTING	GSI	TO BE DET.	05.4.

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
099	RCS/RHR SUCTION LINE INTERLOCKS ON PWRS	GSI	HIGH	05.4.7
100	OTSG LEVEL	GSI	TO BE DET.	NA
101	BWR WATER LEVEL REDUNDANCY	GSI	HIGH	NA
102	HUMAN ERROR IN EVENTS INVOLVING WRONG UNIT OR WRONG TRAIN	GSI	NEARLY RES	18
103	DESIGN FOR PROBABLE MAXIMUM PRECIPATATION	GSI	NEARLY RES	02
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.	06
107	GENERIC IMPLICATIONS OF MAIN TRANSFORMER FAILURES	GSI	TO BE DET.	08
109	REACTOR VESSEL CLOSURE FAILURE	GSI	TO BE DET.	15.
110	EQUIPMENT PROTECTION DEVICES ON ENGINEERED SAFETY FEATURES	GSI	TO BE DET.	06.
113	QUALIFICATION TESTING OF LARGE BORE HYDRAULIC SNUBBERS	GSI	TO BE DET.	03.
115	ENHANCEMENT OF THE RELIABILITY OF THE WEST. SSPS	GSI	HIGH	07

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
116	ACCIDENT MANAGEMENT	GSI	TO BE DET.	018
117	ALLOWABE OUTAGE TIMES FOR DIVERSE SIMULTANEOUS EQUIPMENT OUTAGES	GSI	TO BE DET.	07 OR 16
118	TENDON ANCHORAGE FAILURE	GSI	TO BE DET.	03.8
120	ON-LINE TESTABILITY OF PROTECTION SYSTEMS	GSI	TO BE DET.	03.9.6
121	HYDROGEN CONTROL FOR LARGE DRY PWR CONTAINMENTS	GSI	HIGH	06.2
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	09
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.
125.I.3	SPDS AVAILABILITY	GSI	TO BE DET.	18.2
125.I.4	LONG TERM GEN. ACTIONS - DAVIS BESSE EVENT- PLANT SPECIFIC SIMULATOR	GSI	DROP	18
125.I.5	SAFETY SYSTEM TESTED IN ALL CONDITIONS REQUIRED BY DESIGN BASIS ANALYSIS	GSI	TO BE DET.	07 OR

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.I.6	VALVE TORQUE LIMIT AND BYPASS SWITCH SETTINGS	GSI	TO BE DET.	03.9.6
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.	13
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.	05
125.II.5	THERMAL-HYDRAULIC EFFECTS-LOSS AND RESTORATION OF FDW ON PRIMARY SYSTEM COMPONENTS	GSI	??	05
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.	05.2
127	TESTING AND MAINTENANCE OF MANUAL VALVES IN SAFETY RELATED SYSTEMS	GSI	TO BE DET.	03.9.[illegible]
128	ELECTRICAL POWER RELIABILITY	GSI	HIGH	08 OR APPDX

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
129	VALVE INTERLOCKS TO PREVENT VESSEL DRAINAGE DURING SHUTDOWN COOLING	GSI	TO BE DET.	05.4.7
130	ESSENTIAL SERVICE WATER PUMP FAILURES AT MULTIPLANT SITES	GSI	HIGH	06
131	POTENTIAL SEISMIC INTERACTION INVOLVING THE MOVABLE INCORE FLUX MAP SYSTEM AT WESTINGHOUSE PLANTS	GSI	TO BE DET.	NA
132	RHR PUMPS INSIDE CONTAIMENT	GSI	TO BE DET.	05.4.7
134	DEGREE AND EXPERIENCE REQ. FOR SENIOR OPERATORS	GSI	HIGH	13
135	INTEGRATED STEAM GENERATOR ISSUES	GSI	TO BE DET.	05.4.2
136	STORAGE AND USE OF LARGE QUANTITIES OF CRYOGENIC COMBUSTIBLES	GSI	TO BE DET.	06
137	REFUELING CAVITY SEAL FAILURES	GSI	TO BE DET.	09.1
138	DEINERTING UPON DISCOVERY OF RCS LEAKAGE	GSI	TO BE DET.	05.[illegible]
139	THINNING OF CARBON STEEL PIPING IN LWRS	GSI	TO BE DET.	05.08
140	FISSION PRODUCT REMOVAL BY CONTAINMENT SPRAYS OR POOLS	GSI	TO BE DET.	06
A-01	WATER HAMMER	USI	USI	05.4

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-02	ASYMETRIC BLOWDOWN LOADS ON RCS	USI	USI	15.6
A-03	WESTINGHOUSE STEAM GENERATOR TUBE INTEGRITY	USI	USI	05
A-04	C-E STEAM GENERATOR TUBE INTEGRITY	USI	USI	05.4.2
A-05	B&W STEAM GENERATOR TUBE INTEGRITY	USI	USI	NA
A-09	ATWS	USI	USI	15.8
A-10	BWR FEEDWATER NOZZLE CRACKING	USI	USI	NA
A-11	REACTOR VESSEL MATERIAL TOUGHNESS	USI	USI	05.3
A-12	FRACTURE TOUGHNESS OF S.G. AND RCP SUPPORTS	USI	USI	05.4.14
A-17	SYSTEMS INTERACTION	USI	USI	05 & 15
A-19	DIGITAL COMPUTER PROTECTION SYSTEM	GSI	TO BE REP.	07.2
A-24	QUALIFICATION OF CLASS 1E SAFETY RELATED EQUIPMENT	USI	USI	07.1
A-26	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION	USI	USI	05.3
A-29	PLANT DESIGN FOR REDUCT. OF VULNER. TO SABOTAGE	GSI	MEDIUM	02
A-31	RHR SHUTDOWN REQUIREMENTS	USI	USI	05.4
A-36	CONTROL OF HEAVY LOADS NEAR SPENT FUEL	USI	USI	09

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-39	DETERMINATION OF SAFETY RELIEF VLV POOL DYN LOADS	USI	USI	NA
A-40	SEISMIC DESIGN--SHORT TERM PROGRAM	USI	USI	03.7
A-41	LONG TERM SEISMIC PROGRAM	USI	USI	03.7
A-42	PIPE CRACKS IN BOILING WATER REACTORS	USI	USI	NA
A-43	CONTAINMENT EMERGENCY SUMP PERFORMANCE	USI	USI	06.2
A-44	STATION BLACKOUT	USI	USI	15.3
A-45	SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS	USI	USI	05.4.7
A-46	SEISMIC QUAL. OF EQUIPMENT IN OPERATING PLANTS	USI	USI	NA
A-47	SAFETY IMPLICATIONS OF CONTROL SYSTEMS	USI	USI	05.3
A-48	HYDROGEN CNTRL MEASURES&EFFECTS OF HYDROGEN BURNS	USI	USI	06.2
A-49	PRESSURIZED THERMAL SHOCK	USI	USI	07.1
B-05	DUCTILITY OF TWO-WAY SLABS AND SHELLS -STEEL CONTM	GSI	MEDIUM	03.8
B-06	LOAD, LOAD COMBINATIONS, STRESS LIMITS	GSI	HIGH	03.9.1
B-10	BEHAVIOR OF BWR MARK III CONTAINMENTS	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
B-17	CRITERIA FOR SAFETY RELATED ACTIONS	GSI	HIGH	15 OR 18
B-19	THERMAL-HYDRAULIC STABILITY	GSI	NEARLY RES	04.4
B-22	LWR FUEL	GSI	TO BE REP.	04.2
B-26	STRUCTURAL INTEGRITY OF CONTAINMENT PENETRATIONS	GSI	MEDIUM	06.2
B-29	EFFECTIVENESS OF ULTIMATE HEAT SINKS	GSI	TO BE REP.	06 OR 09
B-31	DAM FAILURE MODEL	GSI	TO BE REP.	02.4.2
B-32	ICE EFFECTS ON SAFETY RELATED WATER SUPPLIES	GSI	TO BE REP.	09
B-53	LOAD BREAK SWITCH	GSI	NEARLY RES	08
B-54	ICE CONDENSER CONTAINMENTS	GSI	MEDIUM	NA
B-55	IMPROVE RELIABILITY OF TARGET ROCK SAFETY RELIEF VALVES	GSI	MEDIUM	05
B-56	DIESEL GENERATOR RELIABILITY	GSI	HIGH	08 & APP
B-58	PASSIVE MECHANICAL FAILURES	GSI	MEDIUM	03 OR
B-60	LOOSE PARTS MONITORING SYSTEM	GSI	NEARLY RES	05
B-61	ALLOWABLE ECCS EQUIPMENT OUTAGE PERIODS	GSI	MEDIUM	06 OR
B-64	DECOMMISSIONING OF REACTORS	GSI	NEARLY RES	05

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
C-08	MAIN STEAM LINE ISOLATION VALVE LEAKAGE CNTRL SYS.	GSI	HIGH	NA
C-09	RHR HEAT EXCHANGER TUBE FAILURES	GSI	TO BE REP.	05.4.7
C-11	ASSESSMENT OF FAILURE AND RELIABILITY OF PUMPS AND VALVES	GSI	MEDIUM	03.9.6
C-14	STORM SURGE MODES FOR COASTAL SITES	GSI	TO BE DET.	02
D-02	EMERGENCY CORE COOLING SYSTEM CAPABILITY FOR FUTURE PLANTS	GSI	TO BE REP.	06.3
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 1.3.4a	HUMAN FACTORS PROGRAM PLAN - MAN MACHINE INTERFACE - LOCAL CONTROL STATIONS	GSI	HIGH	18
HF 2.1	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	

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
HF 5.2	REVIEW CRITERIA FOR HF ASPECTS OF ADVANCED I&C	GSI	HIGH	18
HF 8	MAINTENANCE AND SURVEILLANCE PROGRAM	GSI	HIGH	18
HF.1.3.4b	HUMAN FACTORS PROGRAM PLAN - MAN MACHINE INTERFACE - ANNUNCIATORS	GSI		18
HF.1.3.4c	HUMAN FACTORS PROGRAM PLAN - MAN MACHINE INTERFACE - OPERATIONAL AIDS	GSI		18
HF.1.3.4d	HUMAN FACTORS PROGRAM PLAN - MAN MACHINE INTERFACE - AUTOMATION AND ARTIFICIAL INTELLIGENCE	GSI		18
HF.1.3.4e	HUMAN FACTORS PROGRAM PLAN - MAN MACHINE INTERFACE - COMPUTERS AND COMPUTER DISPLAYS	GSI		18
I.A.1.4	LONG TERM UPGRADING OF OPERATING PERSONNEL	GSI	NEARLY RES	13
I.A.2.2	TRAINING AND QUALIFICATIONS OF OPERATING PERSONNEL	GSI	HIGH	13
I.A.2.6(1)	REVISE REGULATORY GUIDE 1.8	GSI	HIGH	13
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	13

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.3.4	LICENSING OF ADDITONAL OPERATOR PERSONNEL	GSI	MEDIUM	13
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	07 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.[illegible]
II.A.1	SITING POLICY REFORMULATION	GSI	MEDIUM	02
II.B.5	EFFECT OF H2 BURNING AND EXPLOSIONS ON CONT STRUCT	GSI	MEDIUM	06.[illegible]
II.B.6	RISK REDUCTION FOR OPERATING REACTORS WITH SITES WITH HIGH POPULATION DENSITIES	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
II.B.8	RULEMAKING PROCEEDINGS ON DEGRADED CORE ACCIDENTS-HYDROGEN RULE, SEVERE ACCEDENT, ETC.	GSI	HIGH	06.2.5
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	06 & 15.6
II.E.4.3	CONTAINMENT INTEGRITY CHECK	TMI	HIGH	06 & 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	05 & 1[illegible]
II.F.5	CLASSIFICATION OF I & C, AND ELECTRICAL EQUIPMENT	GSI	MEDIUM	07.1
II.H.2	OBTAIN DATA ON INSIDE COND. OF TMI CONTAINMENT	TMI	HIGH	06
II.J.4.1	REVISE DEFICIENCY REPORT REQUIREMENTS	GSI	NEARLY RES	NA
III.A.1.3	(2) MAINTAIN SUPPLIES OF THRYOID BLOCKING AGENT	GSI	NEARLY RES	NA
III.A.3.4	NUCLEAR DATA LINK	GSI	MEDIUM	07.1
III.D.1.1	(2) REVIEW INFORMATION ON PROVISIONS FOR LEAK DETECTION	GSI	TO BE REP.	05.2

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
III.D.1 1	(3) DEVELOP PROPOSED SYSTEM ACCEPTANCE CRITERIA	GSI	TO BE REP.	05.2
III.D.2.3	(4) SUMMARY ASSESSMENT OF LIQUID PATHWAY CONSEQUENCES	GSI	NEARLY RES	11.1
III.D.2.3	(3) LIQUID PATHWAY INTERDICTION	GSI	NEARLY RES	11.1
III.D.2.3	(1) LIQUID PATHWAY RADIOLOGICAL CONTROL	GSI	NEARLY RES	11.1
III.D.2.3	(2) SCREENING OF SITES FOR LIQUID PATHWAY CONSEQUENCES	GSI	NEARLY RES	11.1
III.D.2.5	OFFSITE DOSE CALCULATIONAL MANUAL	GSI	NEARLY RES	15
III.D.3.1	RADIATION PROTECTION PLANS	GSI	HIGH	12
IV.E.5	ASSESS CURRENTLY OPERATING PLANTS	GSI	HIGH	NA
III.D.1.1	(1) PRIMARY CONTAINMENT SOURCES OUTSIDE THE CONTAINMENT STRUCTURE	GSI		05.2

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 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.

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 the overall 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 standardized design. These models will be used to establish a baseline core damage frequency and to determine the dominant core damage contributors for the current System 80 design. In this phase, the System 80 design will be evaluated using generic reliability data.

The second phase will be an interactive process in which these models will be modified to reflect system design enhancements proposed for the System 80+ Standard Design. The models will be evaluated to determine the impact of the design enhancements on core damage frequency and dominant core damage contributors. These impacts will be reviewed and other design enhancements will be considered as appropriate to achieve the overall safety goals.

Phase One: Baseline System 80 PRA

The baseline System 80 core damage frequency calculation is a Level I PRA. 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 Combustion Engineering 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 a valuable method 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 prevent the design from meeting the goals. The System 80+ Standard Design development effort will then be able to focus on improving the reliability of systems or equipment involved in the dominant sequences. As design improvements are adopted, the PRA models will be updated.

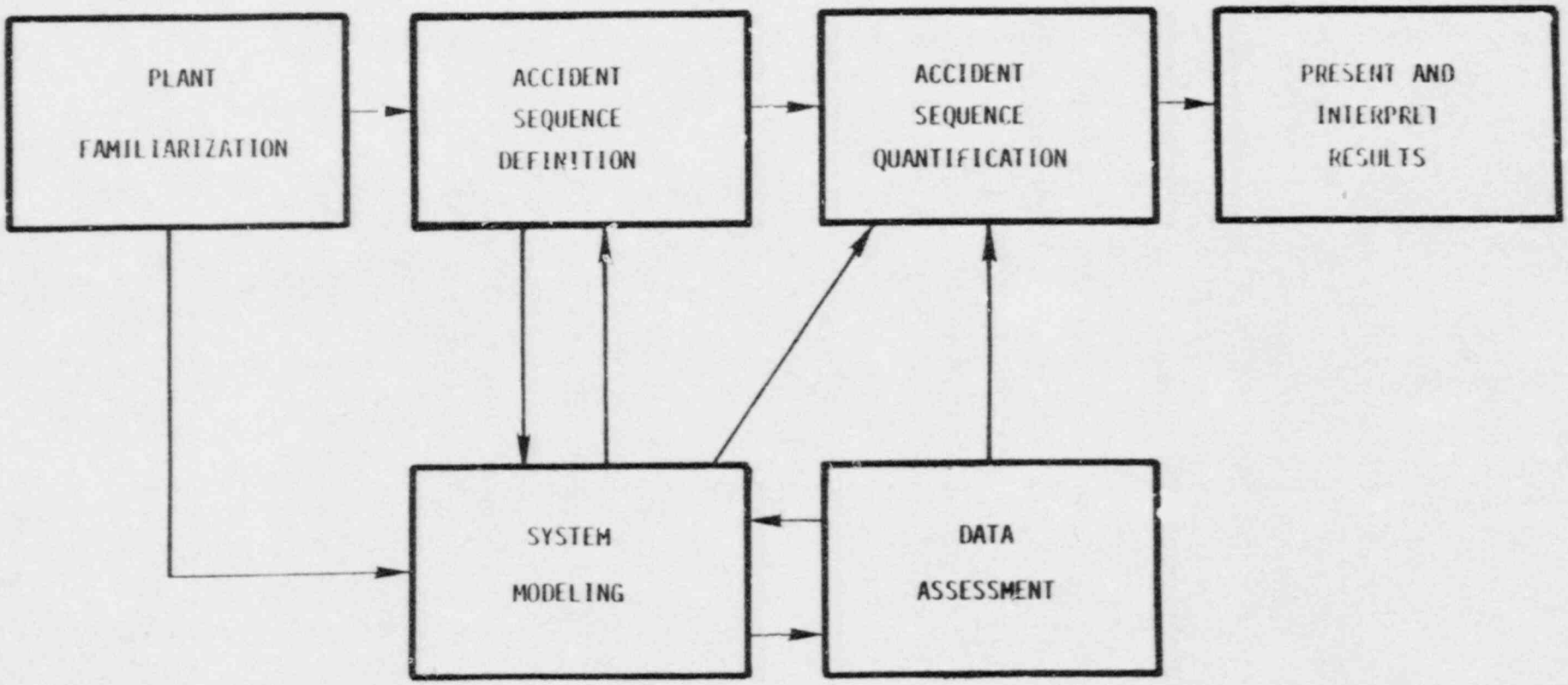
The final PRA for the System 80+ Standard Design will include all of the design modifications that are implemented as a part of the ALWR Design Verification Program. Additionally, with support from the DOE Advanced Reactor Severe Accident Program (ARSAP), the PRA will be upgraded to a Level III PRA and 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 the Calvert Cliffs IREP Report as well as those in NUREG/CR-4550 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 offsite 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 releases 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 of 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 Severe Accident Policy Statement 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 prepare a resolution to each issue. 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 all available 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 issues for the System 80+ Standard Design will be substantiated, as required, by plant specific evaluations. 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 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 a cut-off frequency of 1×10^{-8} per reactor year will not be analyzed. MAAP 3B will be utilized for design-specific analyses of accident initiation, progression, and containment response. It 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 the mitigation of degraded core accidents.

It should be emphasized here that NRC approval of the MAAP code is not required. Technical disagreements between the MAAP 3B results and NRC methods will be addressed on a case-by-case basis in accordance with the review procedures outlined in Chapter 5 of the Licensing Review Bases.

Table C-1

Listing of Planned ARSAP Topic Papers

Set 1 RESOLVED IDCOR/NRC ISSUE - 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 in primary system (IDCOR Issue 4)
- 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 Revaporization of fission product (IDCOR Issue 11)
- o Secondary containment performance (IDCOR Issue 16)
- 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 Debris Coolability (IDCOR Issue 10)

Set 3 PROBABILISTIC METHODS

- o External events
- o Human reliability analysis
- o Success criteria and mission time
- o Common cause failures
- o Accident sequence selection

Set 4 SEVERE ACCIDENT PERFORMANCE

- o Essential equipment performance (IDCOR Issue 18)
- o Criteria for safe stable states

Set 5 SAFETY GOAL EVALUATION

- 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 analysis
- o MAAP acceptance - concensus on severe accident analysis capability

Set 6 SEVERE ACCIDENT MANAGEMENT

- o Severe accident management - planning
- o Severe accident management - equipment capability and operational requirements
- o Effect of plant size on inherent severe accident capability - core power density, fission product inventory, passive heat sinks

FIGURE C-1

THE SEVERE ACCIDENT RESOLUTION PROCESS

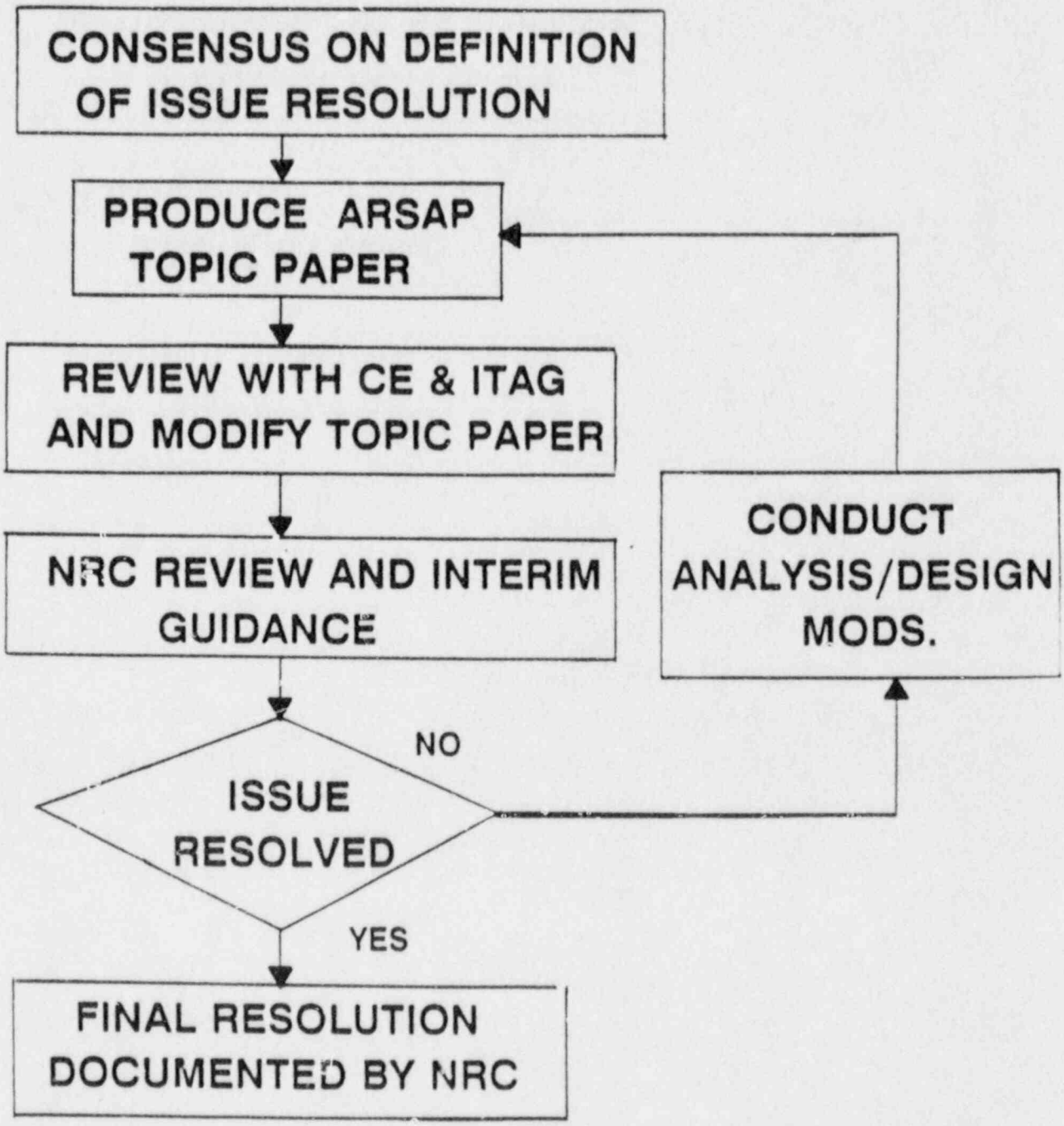
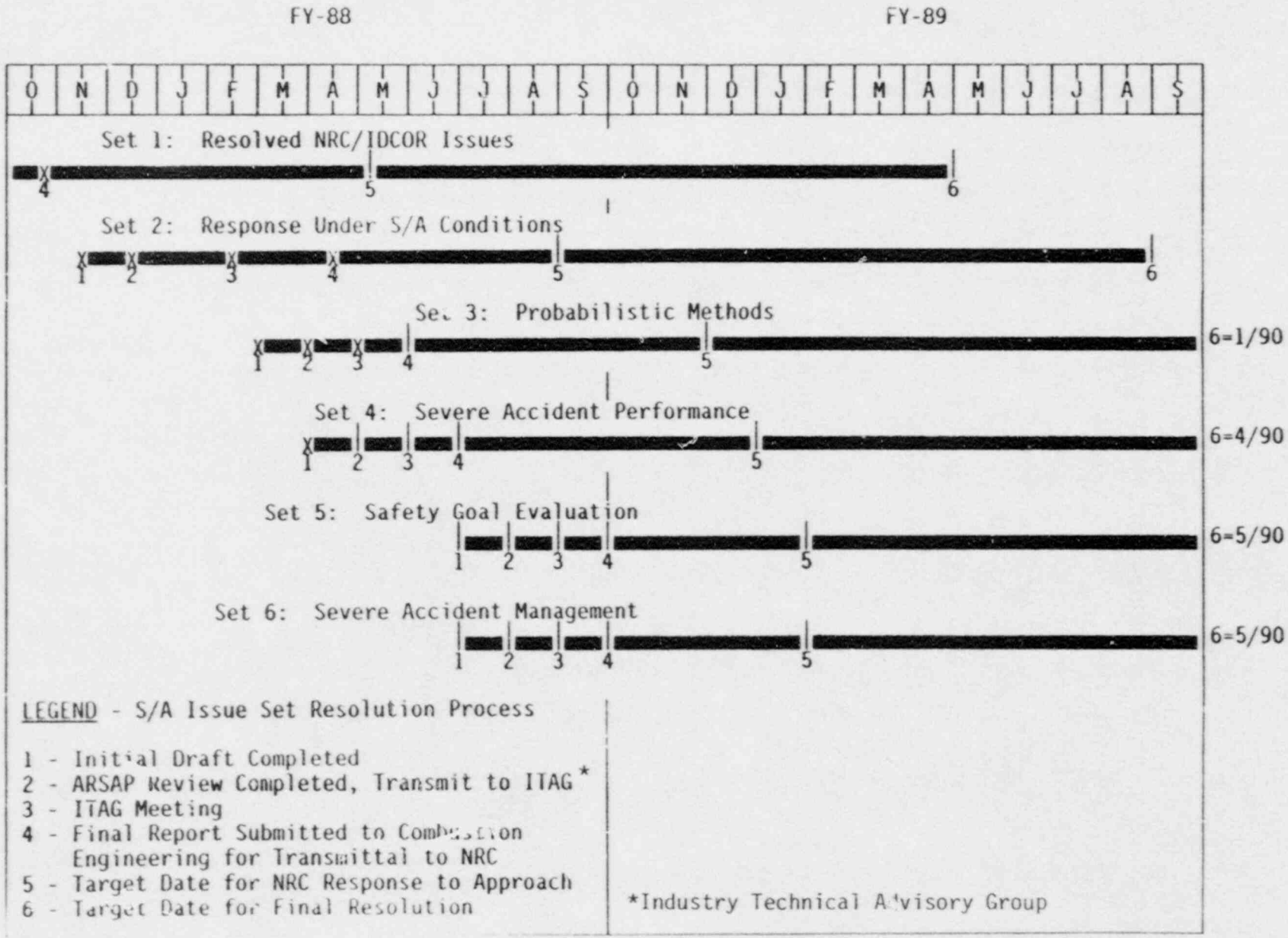


Figure C-2

Identification and Resolution of Severe Accidents



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 Severe Accident Policy Statement and with the NRC Safety Goal Policy Statement. The Safety Goal Policy Statement 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
and
Human Factors Engineering

1.0 INTRODUCTION

The Instrumentation and Controls (I&C) Systems of the System 80+™ Standard Design are integrated into the Nuplex 80+™ Advanced Control Complex (ACC). The purpose of this Appendix is to identify the design standards, criteria and Human Factors Engineering process to be used by Combustion Engineering in the Nuplex 80+ ACC design and to describe the documentation of the design that will be provided to the staff for the Design Certification process.

Combustion Engineering is committed to the I&C standards and criteria currently specified in the base System 80® design (as described in CESSAR F) for which a Final Design Approval (FDA) has been issued. This Appendix will address only those I&C features, requirements and documentation for the System 80+ Standard Design which are different from those as described in CESSAR-F. The differences result from:

(1) Implementation of the Nuplex 80+ ACC

(2) Implementation of EPRI ALWR design requirements that significantly change the I&C design, or

(3) Implementation of Human Factors Engineering in the design of the Control Room.

This Appendix is divided into two principal parts. The first part (Section 2.0) identifies the documentation and criteria applicable to CESSAR-DC Chapter 7 "Instrumentation and Controls." The second part (Section 3.0) addresses the CESSAR-DC Chapter 18 "Human Factors Engineering" documentation required by the SRP to support the I&C systems of the System 80+ Standard Design.

2.0 CESSAR-DC Chapter 7 "Instrumentation and Controls" Documentation

The primary impact the System 80+ design enhancements will have on Chapter 7 is the inclusion of the Nuplex 80+ ACC as part of the Nuclear Power Module. The Nuplex 80+ ACC design includes the following: Control Panels and Workstations, Information Processing and Display Systems, Protection Systems, Control Systems, and Process Instrumentation Monitoring. A revised discussion of these design changes has been included in Amendment B to Chapter 1 of CESSAR-DC (Sections 1.2.5 and 1.2.6).

The instrumentation and control systems of the System 80+ Standard Design are to use fiber optics, multiplexing, and computer controls. Although Staff guidance in this area has not been developed, Combustion Engineering has committed to the standards and criteria currently specified in the SRP and has previously licensed safety related systems that utilize these technologies. The descriptions of these I&C design changes and expanded scope items will be completed within CESSAR-DC to a level of detail similar to that found in CESSAR-F and other applicant's FSARs (for those portions previously not within the CESSAR scope). Specifically, Combustion Engineering will provide sufficient information to allow the Staff to determine the acceptability of the I&C systems for the System 80+ Standard Design.

In lieu of actual test or qualification reports for equipment that will be selected later by Combustion Engineering or a utility applicant, Combustion Engineering will provide a description of the Verification and Validation (V&V) plan(s) and the testing or qualification standards to be used to assure that the equipment that is ultimately selected will perform as intended. Full V&V of safety related systems using commercially available (off-the-shelf) computer equipment is intended to be imposed at the point in the design and fabrication process where such equipment becomes designated nuclear safety-related.

A list of Standardized Functional Descriptions (SFDs) for the systems required to support the Nuplex 80+ ACC design is provided in Table D-1.

Table D-1

I&C Standardized Functional Descriptions

Control Room

Emergency Operations Facility

Technical Support Center

Electric Power Distribution System

Fire Protection System

Diesel Generator System

Nuclear Service Water System

Component Cooling Water System

Atmospheric Dump Valves

Automatic Dispatch System

Environmental Support Systems (HVAC)

3.0 CESSAR-DC Chapter 18 "Human Factors Engineering" Documentation

The CESSAR-DC Chapter 18 submittal will document the Human Factors Engineering program integrated into the design of the Nuplex 80+ ACC. The Nuplex 80+ ACC is being developed from the Nuplex 80 advanced control room design. The design is being modified to incorporate current Human Factors Engineering principles, meet recent regulatory requirements and address requirements from the EPRI ALWR Requirements Document. This Section will briefly document the design process relative to the Human Factors Engineering considerations and identify the documentation to be provided in Chapter 18 of CESSAR-DC.

3.1 Overview

Human Factors Engineering principles will be incorporated into the Nuplex 80+ ACC in all phases of the design. The design process for the Nuplex 80+ ACC will begin by establishing design goals and design bases for the control center complex. Information and control requirements will then be established by using an independent top-down systems/functions analysis in conjunction with existing System 80+ instrumentation requirements. A configuration evaluation will be conducted to define configuration and workspace criteria, select a panel arrangement and console profiles and establish supporting standardized functional descriptions. A Nuplex 80+ ACC panel layout evaluation will establish information display, alarm and control methodologies and then allocate information and control requirements to the most appropriate method. Panel layout criteria will be defined and applied to develop actual panel layouts. The panel design will be carried out in complete detail for one Nuplex 80+ ACC panel: the Reactor Coolant System (RCS). Representative layouts of all other Nuplex 80+ ACC panels will be developed by a similar method.

The design of control and monitoring facilities outside the main control room will be addressed to ensure all appropriate interfaces with the NUPLEX 80+ ACC are addressed (Section 18.4.4). This will include the Remote Shutdown Panel (RSP), Technical Support Center (TSC), and Emergency Operations Facility (EOF). A verification and validation process applicable to the complete Nuplex 80+ ACC design will be documented.

Table D-2 presents a draft outline for Chapter 18 of CESSAR-DC and provides a list of deliverables to be provided in each section. The following paragraphs will describe briefly the intended design process resulting in these deliverables.

3.2 Design Process

The Nuplex 80+ ACC will be designed by a multi-disciplinary team whose organization and responsibilities will be documented in CESSAR-DC, Section 18.1. The initial design effort will be to establish and document design goals and design bases for the Nuplex 80+ ACC. These will be documented in Section 18.2. Section 18.3 will be dedicated to the design process and the application of Human Factors Engineering in that process. Section 18.4 will document the evaluation results and the Nuplex 80+ ACC design for each of the following design phases.

The Nuplex 80+ ACC information and control requirements will be developed using the following approach. An independent top-down systems/ function analysis will be performed and the System 80 instrumenta[illegible] requirements specified for existing plants will be obtained. The approach will be to specify detailed information and control requirements only for those tasks and functions related to the RCS for subsequent use in the detailed evaluat[illegible] [illegible]nel la[illegible] (see Table D-2). The top-down analysis will establis[illegible] [illegible]ational [illegible]es in all modes of operation (emergency and normal) [illegible] [illegible]nal analysis for all plant systems. A man-mac[illegible] [illegible] tion will be performed and tasks will be specified [illegible] [illegible] functions relevant to the RCS. A current i[illegible]ment list fo[illegible]ng System 80 plants will be used as a starting

point for information and control requirements for all other systems. This will be modified to include System 80+ improvements and a representative BOP design. This list will be used for developing panel layouts for all Nuplex 80+ ACC panels other than the RCS panel. A comparison will be made between the experience-based instrument list and the results of the system/function analysis for the RCS. The documentation to be provided in Chapter 18 for the information and control requirements is listed in Sections 18.3 and 18.4 of Table D-2. A complete systems/functions task analysis will be performed later by the same method as documented in CESSAR-DC.

The Nuplex 80+ control room configuration evaluation will first establish the operational requirements for the Nuplex 80+ ACC. Human Factors Engineering criteria for configuration and workspace will be defined and candidate designs evaluated to select a final configuration. Panel arrangements and console profiles will be established based on Human Factors Engineering studies and the functional analysis performed in the previous effort. Criteria for the control room environment, communications, and other interfaces will be defined and will be used to generate appropriate standardized functional descriptions. Deliverables from this effort are listed in Sections 18.3 and 18.4 of Table D-2.

Nuplex 80+ ACC panel layouts will be developed for all control room panels. A detailed panel layout evaluation will be conducted for the RCS panel. This will document the complete panel layout design process and results. Panel layouts for all other panels will be completed by a similar method based on less design detail.

Man-machine interface design bases will be developed and information display, alarm and control methods will be established. These will be documented in detail in Chapter 18. The detailed panel layout evaluation for the RCS will use the task analysis results as a basis for its information and control requirements. Information display allocation criteria will be developed and the information and control requirements allocated to the appropriate methods. Human Factors Engineering

panel layout criteria will be established and the design of the RCS panel will be completed. Both the information display allocation and control panel layouts will make extensive use of the results of the top-down systems/functions analysis. Complete detail of alarm grouping and logic, validation logic, indicator parameter groupings and associated CRT displays will be provided. A list of documentation to be provided in Chapter 18 for the RCS panel layout is given in Table D-2.

The remaining Nuplex 80+ ACC panels will be developed with a similar method, although with less design detail, using the System 80+ Standard Design instrument list and a representative BOP design as the basis for information and control requirements. The panel layouts will be developed according to the same criteria used for the RCS panel. A list of documentation provided for these Nuplex 80+ panels is given in Table D-2.

The final aspect of the Nuplex 80+ ACC design process will focus on the RSP, TSC and EOF. For the RSP, information and control requirements will be developed, Human Factors Engineering criteria will be established and a panel layout will be provided. For the TSC and EOF only human engineering criteria related to Nuplex 80+ ACC displays for these work-stations will be provided. The documentation for these Human Factors Engineering efforts will also be in Sections 18.3 and 18.4, as shown in Table D-2.

The verification and validation process for the main control room and other workstations will be defined in detail in Chapter 18. This will include the method to be used for verifying task performance capabilities primarily based on the complete system/function task analysis. The validation process will demonstrate that all control room functions can be accomplished and that the Human Factors Engineered man-machine interface is adequate.

The physical design of control room panels and workstation(s) shall be in accordance with General Design Criterion 19, IEEE-323 and IEEE-344, Regulatory Guide 1.75 and Branch Technical Position CMEB 9.5-1 and other applicable standards.

Guidance for establishing control room panel conformance to separation, isolation, fire protection and seismic criteria will be provided within Chapter 18.

Table D-2
Chapter 18 - Draft Outline

18.1 Design Team Organization and Responsibilities
18.2 Design Goals and Design Bases
18.3 Design Process and Application of Human Factors Engineering

- 18.3.1 Information and Control Requirements
- 18.3.2 Control Room Configuration Assessment
- 18.3.3 Panel Layout Evaluation
- 18.3.4 Control and Monitoring Stations Outside the Main Control Room
- 18.3.5 Verification and Validation Process

18.4 Nuplex 80+ Control Complex Design Analyses

18.4.1 Information and Control Requirements
- Systems/Functions and Operational Sequences List
- Man-Machine Functional Allocation
- Task List for RCS-Related Functions, Including Categorization by Functions and Operational Sequences
- Comparison of System 80+ Experienced-based Information and Control Requirements with Function and Task Lists
- Final List of Nuplex 80+ Information and Control Requirements for All Systems

18.4.2 Control Room Configuration Assessment
- Operational Requirements
- Human Factors Engineering Criteria for Configuration and Workspace (from NUREG 0700, NP-3659, etc.)
- Results of Configuration Evaluation
- Documentation of Nuplex 80+ Configuration, Including:
  - o Panel Arrangement and Configuration Dimensions
  - o Results of Workspace Studies
    - Visibility
    - Mobility
    - Access
    - Operator Furnishings
  - o Console Profiles
  - o Anthropometric and Ergonomic Study Results
  - o Allocation of Sit Down/Stand Up Panels
- Control Room Interface (Environmental, Communication, Habitability, etc.) Criteria List

18.4.3 Panel Layout Evaluation
- Man-Machine Interface Design Bases List
- Detailed Descriptions of Information Display and Control Methods, Including Characteristics of Presentation Techniques, Operator Interaction, Relationship to Other Panel Display Methods, and Failure Modes
- Information/Display and Control Allocation Criteria List

Table D-2 (Cont'd)
Chapter 18 - Draft Outline

- Results of Allocation Criteria Application
- Human Factors Engineering Panel Layout Criteria, Including CRT Display Criteria
- Panel Layouts and CRT Displays for the RCS Panel, Including:
  - o Panels
    - Panel Drawings
    - Alarm Windows List
    - Alarm Points for Grouped Alarms
    - Discrete Indicators List with All Parameters, Channels, and Logic Provided
      - Operator Module Contents
    - Process Controls List
    - Discrete Controls List and Subgroup Control Allocation
  - o CRT Displays
    - Content and Layout of System Displays for Selected Systems
    - Content and Layout of Alarm Displays for Selected Systems
    - Integrated Process Status Overview Display
    - Application Program Displays
- Panel Layouts for All Other Nuplex 80+ Panels, Including:
  - o Panel Drawings
  - o Representative Alarm Windows List and Grouped Alarms
  - o Representative Discrete Indicators Parameter List
  - o Operator Modules
  - o Representative Process Controls List
  - o Representative Discrete Controls List and Subgroup Control Allocation

18.4.4 Control and Monitoring Stations Outside the Main Control Room

- Remote Shutdown Panel
  - o Information and Controls Requirements
  - o Human Engineering Criteria (Panel Layout, Environmental)
  - o Panel Layout
- EOF Human Engineering Criteria List
- TSC Human Engineering Criteria List