NUREG-0852

Safety Evaluation Report

related to the final design of the Standard Nuclear Steam Supply Reference System CESSAR System 80

Docket No. STN 50-470

Combustion Engineering, Incorporated

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

November 1981

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SAFETY EVALUATION REPORT RELATED TO
THE FINAL DESIGN APPROVAL OF THE
COMBUSTION ENGINEERING STANDARD
NUCLEAR STEAM SUPPLY SYSTEM
(CESSAR)

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This is a safety evaluation report (SER) on the application for a Final Design Approval (FDA) for the Combustion Engineering Standard Safety Analysis Report (CESSAR) based upon the application filed by Combustion Engineering, Inc. (CE). This report was prepared by the United States Nuclear Regulatory Commission staff (NRC staff or staff) and summarizes the results of the staff's safety review of the CE standard nuclear steam supply system, which CE has designated "System 80." The term NRC as used in this document refers to the staff's position. The NRC Licensing Project Manager for CESSAR is Mr. Christopher Grimes. Mr. Grimes may be contacted by calling (301) 492-9798 or writing: Division of Licensing, U.S. Nuclear Regulatory Commission, Washington, DC 20555. The NRC staff principal reviewers for this project are listed in Appendix E.

The application for a Preliminary Design Approval (PDA) for CESSAR was filed with the United States Atomic Energy Commission (now the Nuclear Regulatory Commission) on September 17, 1973. Following the review by the staff and the Advisory Committee on Reactor Safeguards (ACRS), PDA-2 was issued to CE on December 31, 1975 in conjunction with the staff's SER, NUREG-75/112. The application for an FDA was filed on October 27, 1978 and was subsequently docketed for review on December 21, 1979.

Before issuing an FDA for a standard nuclear steam supply system, the NRC staff is required to review the effects of the system design and its interfaces with the balance-of-plant (BOP) on the public health and safety. The safety review of CESSAR has been conducted in accordance with the Commission's standardization policy as published in the FEDERAL REGISTER (42 FR 34395 and 43 FR 38954), and is based on the Final Safety Analysis Report (FSAR) that accompanied the FDA application and Amendments 1 through 5 thereto. During the course of the review the staff has met a number of times with CE, their subcontractors, and the CESSAR reference-plant utilities to discuss the design and interfacing of CESSAR. As a result, additional information was requested, which CE provided in Amendments 1 through 5 to the FSAR and letter and topical reports. All of this information is available to the public for review at the NRC Public Document Room at 1717 H Street NW., Washington, DC.

Following the Three Mile Island Unit 2 (TMI-2) accident, the Commission paused in its licensing activities to assess the impact of the accident. During this pause the recommendations of several groups established to investigate the lessons learned from TMI-2 became available. All available recommendations were correlated and assimilated into a "TMI Action Plan" now published as NUREG-0660, entitled "NRC Action Plan Developed as a Result of the TMI-2 Accident." Additional guidance relating to implementation of the Action Plan is given in NUREG-0737, "Clarification of TMI Action Plan Requirements." These licensing requirements have been established to ensure incorporation of the lessons learned from the TMI-2 accident to provide additional safety margins.

The regulations governing the submittal and review of standard designs under the "reference system" option are contained in paragraph 2.110 of 10 CFR Part 2 and Appendix O to 10 CFR Part 50. CE is responsible for the design of those systems within the CESSAR System 80 scope and sufficiently descriptive interface information such that applicants for an operating license (OL) referencing CESSAR need only describe the BOP systems, site-related information, and conformance with the CESSAR interface requirements. During the course of the review, the staff audited detailed design analyses and interface documentation which are retained in CE's offices in Windsor, Connecticut. Summaries of this information are described in the CESSAR FSAR and are discussed in the text of this report.

Sections 2 through 21 of this report address the NRC review and evaluation of non-TMI-related requirements that have been considered during the course of the staff's review of the application for an FDA for CESSAR, and generally follows the format of the Standard Review Plan, NUREG-0800. Section 22 of this report contains the NRC review and evaluation of CE's response to the TMI-related requirements. The summary conclusions of this report are presented in Section 23.

Appendix A is a chronology of NRC's principal actions related to the reivew of the application. Appendix B is a bibliography of the references used during the course of the review. Appendix C is a discussion of how various ACRS (Advisory Committee on Reactor Safeguards) generic concerns relate to the CESSAR application. Appendix D is a list of abbreviations used in this report. Appendix E is a list of principal contributors.

The review and evaluation of CESSAR for an FDA is only one stage in the continuing review by the staff of the design, construction, and operation of facilities that reference CESSAR. The proposed CESSAR design was reviewed as part of the PDA application. Construction of the facilities referencing CESSAR is being monitored in accordance with the inspection program of the staff. At this, the FDA review stage, the NRC staff has reviewed the final design to determine that the Commission's safety requirements have been met. If an FDA is granted, applicants may reference CESSAR with regard to the acceptability of the systems within the CESSAR System 80 scope.

1.2 General Description

The CESSAR reference system consists of a pressurized water reactor (PWR) with two primary coolant loops, each containing a steam generator and two reactor coolant pumps, an electrically heated pressurizer vessel connected to one of the loops, and the auxiliary and safety systems directly related to the nuclear steam supply system (NSSS). The NSSS design generates approximately 3800 MWt, producing saturated steam for use in a BOP turbine-generator system.

Although the CESSAR scope does not include the BOP features and systems (e.g., site, plant buildings and structures, ultimate heat sink, onsite and offsite electrical systems, and the turbine-generator and its auxiliary systems), the CESSAR scope does include the minimum interface requirements pertaining to those BOP features that have a direct bearing on the integrity or functional capability of the safety-related systems within the CESSAR scope. The CESSAR design, as depicted in Figure 1.1, consists of the following systems:

(1) Reactor System
(2) Reactor Coolant System
(3) Reactor Control System
(4) Reactor Protective System
(5) Engineered Safety Features Actuation System
(6) Chemical and Volume Control System
(7) Shutdown Cooling System
(8) Safety Injection System
(9) Fuel Handling System

The reactor system consists of the reactor vessel, the reactor core, reactor core, reactor internals and core supports, and the vessel head. The reactor core is composed of 241 fuel assemblies, each having a 16 × 16 array of Zircaloy clad uranium dioxide pellets, and 89 or more control element assemblies, which are clusters of either 4 or 12 boron carbide neutron absorbing rods.

The reactor coolant system consists of two closed reactor coolant loops. Each loop contains a steam generator, two reactor coolant pumps, and a 42-inch ID outlet (hot) pipe and two 30-inch ID inlet (cold) pipes to the reactor vessel. Water, which both moderates and cools the core, will be heated in the reactor vessel and flow through the steam generators where the heat will be transferred to the secondary (steam turbine) system and back to the reactor through the pumps to complete the cycle. An electrically heated pressurizer vessel with a safety valve system connected to the hot leg in one of the loops, which will establish and maintain system pressure.

The reactor power will be controlled by two reactivity control systems: (1) control element assemblies, which will provide changes in reactivity for startup or major changes in power, and (2) dissolved boron in the primary coolant, which will compensate for long-term variations in reactivity due to fuel burnup and the buildup of fission products and will ensure sufficient shutdown margin during refueling. The vertical movement of the control element assemblies will be accomplished by magnetic jack-type drive mechanisms and the boron concentration will be adjusted and controlled by the Chemical and Volume Control System.

The Reactor Protective System (RPS) consists of the sensors, software, logic circuits, and related supporting equipment to monitor selected NSSS conditions which will initiate protective action when any two of the four independent instrumentation channels of a given parameter reach their preset values. The protective action consists of interrupting power to the control element drives such that they drop into the core and shutdown (scram) the reactor. A Supplementary Protection System (SPS) augments the RPS by providing a separate and diverse trip logic which will initiate a reactor shutdown when the pressurizer pressure exceeds the setpoint value.

The Engineered Safety Features Actuation System (ESFAS) within the CESSAR scope consists of the sensors, software and logic circuits which will provide signals

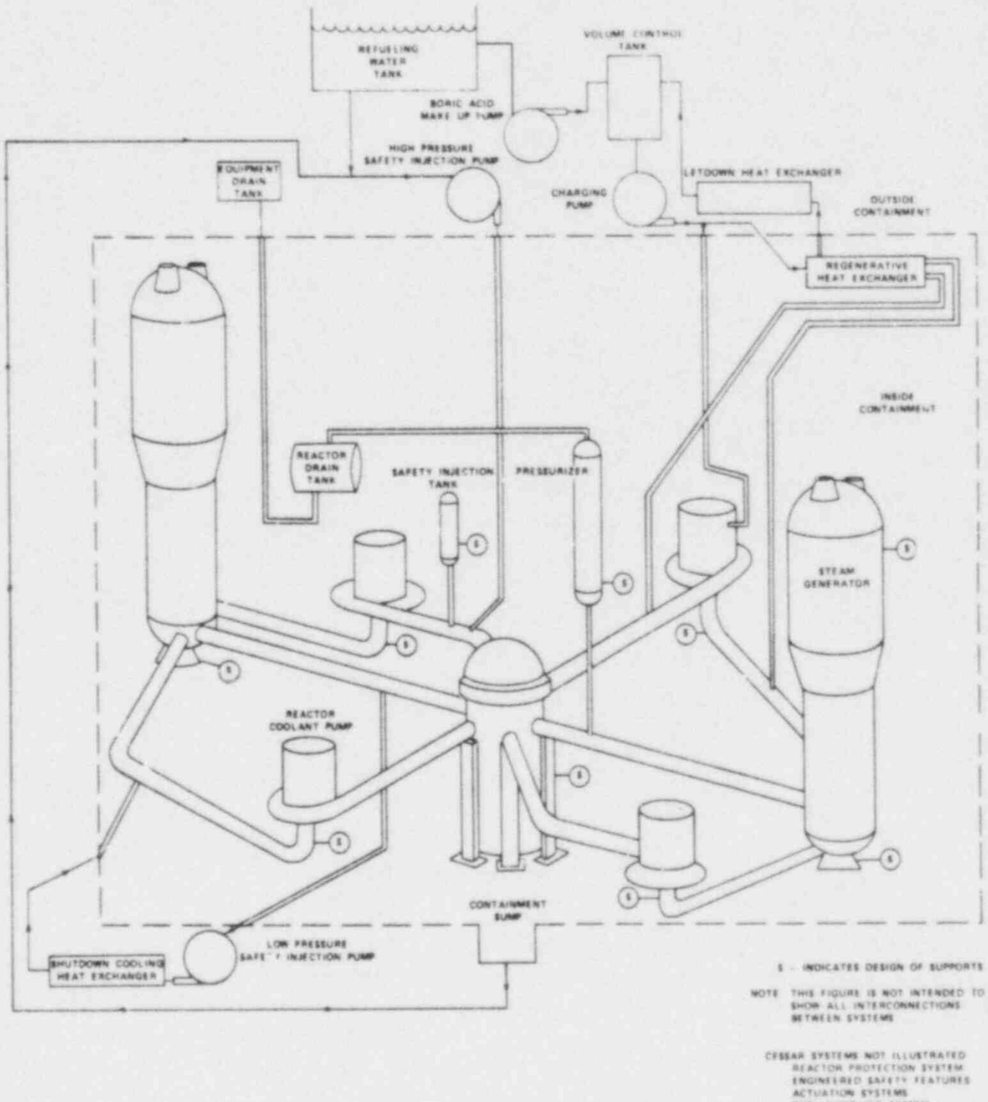

Figure 1.1 - CESSAR Design Scope

to actuate safety-related systems for accident and transient events. The ESFAS interfaces with circuitry and equipment within the BOP scope.

The Chemical and Volume Control System (CVCS) will maintain the purity, volume, and boron concentration of the primary system coolant. Purity is controlled by the continuous purification of a bypass stream of reactor coolant. Volume is controlled by maintaining the level of reactor coolant in the pressurizer by automatically varying the amount of coolant discharged (letdown) and the amount pumped back into the system by the charging pumps. The boron concentration will be controlled by a "feed and bleed" method, where the purified letdown stream will be diverted to a boron recovery section of the CVCS and either concentrated boric acid or demineralized water will be added to the reactor coolant by the charging pumps.

The shutdown cooling system will reduce the reactor coolant temperature from the normal operating temperature of approximately 350 degrees Fahrenheit to a refueling temperature of approximately 135 degrees Fahrenheit. This will be accomplished by using the low pressure safety injection pumps to circulate reactor coolant through the shutdown cooling heat exchangers.

The Safety Injection System (SIS) provides emergency core cooling in order to control, mitigate, and terminate postulated accidents, including a loss-of-coolant accident (LOCA). The SIS includes four safety injection tanks, and independent and redundant low pressure and high pressure safety injection subsystems which will automatically inject highly borated water into each of the four reactor coolant system cold legs. The SIS will provide cooling to limit core damage and fission product releases, and ensure an adequate reactivity shutdown margin.

The fuel handling equipment within the CESSAR scope consists of (1) the refueling machine, (2) the CEA change platform, (3) the fuel transfer system, (4) the spent fuel handling machine, and (5) the new fuel and CEA elevators. The purpose of this equipment is to provide for the safe transfer of new fuel, spent fuel and CEA's between the fuel storage facility, the containment, and the fuel shipping and receiving area. This equipment also provides for the assembly, disassembly, and storage of the reactor vessel head and reactor internals.

1.3 Comparison With Similar Facility Designs

Many features of the design of CESSAR are similar to those the staff has evaluated and approved previously for other PWR plants now under construction or in operation (for example, Waterford, Unit 3; Saint Lucie, Unit 2; San Onofre, Units 2 and 3; and Arkansas Nuclear One, Unit 2). To the extent feasible and appropriate, NRC has relied on earlier reviews for those features that were shown to be substantially the same as those previously considered. Where this has been done, the appropriate sections of this report identify the other facilities involved. SERs for these other facilities have been published and are available for public inspection at the Nuclear Regulatory Commission's Public Document Room at 1717 H Street NW., Washington, DC.

1.4 Identification of Agents and Contractors

Combustion Engineering, Incorporated (CE) is the principal designer and contractor for the CESSAR System 80 design. The standard plant NSSS design has been established such that any architect engineer and utility can reference CESSAR, provided the interface requirements can be satisfied. Those utilities that reference CESSAR and currently have construction permits now in effect are listed in Table 1.4-1.

Table 1.4-1 Current CESSAR reference plants

Plant	Utility
Palo Verde Units 1-3	Arizona Public Service
WPPSS Units 3 & 5	Washington Public Power Supply
Yellow Creek Units 1 & 2	Tennessee Valley Authority
Perkins Units 1-3	Duke Power Company
Cherokee Units 1-3	Duke Power Company

1.5 Summary of Principal Review Matters

NRC technical review and evaluation of the information submitted by the applicant considered, or will consider, the principal matters summarized below:

(1) The design, fabrication, and testing criteria, and expected performance characteristics of the system and components important to safety to determine that they are in accord with the Commission's General Design Criteria (GDC), Quality Assurance Criteria (QAC), Regulatory Guides, and other appropriate rules, codes, and standards, and that any departures from these criteria, codes, and standards, have been identified and justified. In accordance with the Commission's standardization policy (43 FR 38955), CESSAR has been reviewed against the regulatory guidelines in effect at the time staff positions were issued in connection with the review of the PDA and those new safety requirements that the Director of Nuclear Reactor Regulation has directed to be implemented. The new safety requirements are addressed in Appendix A to the CESSAR FSAR and TMI-related requirements are addressed in Appendix B to the CESSAR FSAR.

(2) The expected response of CESSAR to various anticipated operating transients and to a broad spectrum of postulated accidents. Based on this evaluation, NRC determined that the potential consequences of a few highly unlikely postulated accidents (design-basis accidents, DBAs) would exceed those of all other accidents considered. The staff performed conservative analyses of these DBAs to determine that the calculated potential offsite radiation doses that might result, in the very unlikely event of their occurrence,

would not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100 for the CESSAR site envelope.

1.6 Modifications to CESSAR During the Course of NRC Review

During the review, the NRC staff met a number of times (see Appendix A to this report) with CE's representatives, contractors, and consultants to discuss various technical matters related to the CESSAR design. Subsequently, CE made a number of changes to the CESSAR design as a result of the NRC review. The staff reviewed these design changes as well. Special details concerning these changes are included in amendments to the FSAR and in the appropriate subsections of this report.

1.7 Summary of Outstanding Issues

Section 18 is reserved for the report by ACRS to be issued following its review of the CESSAR application and this SER. The ACRS report will be included in a revised version of this SER.

As a result of NRC review of the safety aspects of the CESSAR design, a number of items remain outstanding at the time of issuance of this report. Since the staff has not completed its review and reached final positions in these areas, NRC considers these issues to be open. The review of these items will be completed before issuing an FDA and will be reported in a revised version of this report. The open items, with appropriate references to subsections of this report, are summarized below.

(1) Environmental qualification (3.11)
(2) Fuel rod pressure limits (4.2.1.1(h))
(3) CPC software and schedule (4.4.5, 4.4.11)
(4) ICC instrumentation (22.2, II.F.2)

1.8 Confirmatory Issues

At this point in the review there are a few items that have essentially been resolved to the staff's satisfaction, or can be readily resolved on a plant-specific basis, but for which certain conformatory information has not yet been provided by the CE. In these instances, CE has committed to provide the confirmatory information in the near future. If staff review of the information does not confirm preliminary conclusions, that item will be treated as open and NRC staff will report on its resolution in a revision to this report.

(1) Preoperational vibration testing program (3.9.2)
(2) Pump and valve operability program (3.9.3.1)
(3) Fuel performance analyses (as listed in 4.2.5)
(4) Cladding collapse analysis (4.2.3.2(b))
(5) Supplemental ECCS analysis (4.2.3.2(f))
(6) Partial-loop operation (4.4.9)
(7) Reactor power cutback system (4.4.11, 7.2.1.3)
(8) Operators for 2 SDCS valves (5.4.3)
(9) Shutdown cooling analysis (5.4.3)
(10) Boron mixing testing (5.4.3)
(11) Isolation Valve power (6.4.2)

(12) Containment sprays (6.5)
(13) Boron dilution alarms (15.2.4.5)
(14) Small steam line break analysis (15.3.1)
(15) Feedwater line break analysis (15.3.2)
(16) RCP shaft seisure analysis (15.3.3, 15.4.2)
(17) Steam line break (15.3.1, 15.4.1)
(18) Steam generator tube rupture analysis (15.4.5)
(19) Fuel handling accident analysis (15.4.6)
(20) Effects of loss of AC power on pump seals (22.2, II.K.3.25)

In addition, the staff is continuing to review specific CESSAR interface requirements with regard to their acceptability and completeness for future reference plant applications, as discussed in Sections 1.10, 5.1, 6.3.2, 6.5, 9.3.3, and 9.3.4 of this report.

1.9 Generic Issues

The ACRS periodically issues a report listing various generic matters applicable to light water reactors (LWRs). A discussion of these matters is provided in Appendix C to this report which includes references to sections of this report that more specifically discuss CESSAR concerns.

The NRC continuously evaluates the safety requirements used in its review against new information as it becomes available. In some cases, the staff takes immediate action or interim measures to assure safety. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether existing requirements should be modified. These issues being studied are sometimes called generic safety issues because they are related to a particular class or type of nuclear facility. A discussion of NRC's program to resolve these generic issues is presented in Appendix C to this report.

1.10 Interface Information

CESSAR is a standard nuclear steam supply system. Consequently, the CESSAR FSAR does not describe an entire facility, but is limited in scope to those design and safety features associated with the NSSS and the related standard features of the CESSAR design. The CESSAR design scope is defined in Table 1.10-1. CESSAR also defines interface requirements which must be imposed on the reference plant such that the BOP will provide compatible design features that will ensure the applicability, functional performance, and safe operation of the CESSAR systems.

The interface requirements range in type from general design provisions (e.g., provide protection from potential missiles) to detailed design provisions (e.g., system pressure, temperature, and rated flow). In some cases, the interfaces are in the form of detailed design analyses that are needed by the BOP designer for the analyses of interconnected systems and structures. The staff's review and the evaluation reflected in this report address the interface requirements either from the standpoint of general design provisions (i.e., qualitative); specific design provisions (i.e., quantitative), by incoporation or reference to the interface requirements in the CESSAR FSAR; or, by a description of the interface mechanism between CESSAR and the BOP. A summary of the interface requirements, with references to specific sections of this report, is presented in Table 1.10-2.

During the course of the review, the staff audited the detailed interface information that is supplied to the reference plant applicant. The safety-related interface requirements are described in the CESSAR FSAR and are discussed in this report. This interface information has always been a part of the contractual arrangements between the NSSS designer and the BOP designer (architect-engineer). However, for the purpose of a standard NSSS design, the safety-related interface requirements are significant for future referencability.

In general, we have determined that the interface requirements provided in CESSAR are adequately descriptive to assure the compatibility of the CESSAR System 80 design with the BOP designs that would be submitted in applications referencing CESSAR. The relative acceptability of specific interface requirements is described in the appropriate subsections of this report.

The staff's review of CESSAR was conducted in parallel with the first reference plant application, the Palo Verde Nuclear Generating Station. Although the review of a CESSAR reference plant should address both the BOP design and conformance with the CESSAR interface requirements, this was not necessary in the Palo Verde review since both the standard plant and reference plant designs were reviewed at the same time making the check of the interface requirements unnecessary. The staff is continuing to review specific interface requirements to determine whether they are acceptable and complete for future reference plant applications. We will report on the resolution of this issue in a revision to this report, prior to issuing an FDA for CESSAR. The results of this review may identify additional interface requirements. For future reference plant applications, subject to the resolution of the outstanding issues identified in this report, the scope of the review will be limited to the BOP design and conformance with the CESSAR interface requirements, as supplemented by the interface requirements in this evaluation.

Table 1.10-1 CESSAR Design Scope - Systems and Equipment

1. Reactor Coolant System

   a) Reactor Vessel Assembly

      1. Reactor Vessel Internals
      2. Fuel and Fuel Assemblies
      3. Surveillance Specimens and Holders
      4. Neutron Sources
      5. Control Element Assemblies
      6. Control Element Drive Mechanisms
      7. Reactor Vessel Supports
      8. Closure Studs, Nuts and Washers
      9. Reactor Vessel Head Closure Seal

   b) Steam Generator Assembly

      1. Steam Generator Internals
      2. Steam Generator Supports

   c) Pressurizer Assembly

      1. Pressurizer Heaters
      2. Pressurizer Supports

   d) Reactor Coolant Pumps

      1. Reactor Coolant Pump Supports
      2. Reactor Coolant Pump Instrumentation and Component Controls

   e) Reactor Coolant Piping Including Pipe Stop Weld Buildups
   f) Main Steam and Feedwater System Instrumentation and Component Controls
   g) RCS Instrumentation and Component Controls
   h) Spray Line Valves
   i) Insulation

2. Engineered Safety Features Systems

   a) Safety Injection Tanks

      1. Safety Injection Tanks
      2. High Pressure Safety Injection Pumps
      3. Low Pressure Safety Injection Pumps
      4. Associated Valves
      5. Instrumentation and Component Controls

   b) Containment Isolation System - Isolation Provisions

      1. Safety Injection System High and Low Pressure Injection Lines
      2. Containment Sump Suction Lines

Table 1.10-1 (Continued)

3. Shutdown Cooling Suction Lines
4. Letdown Line
5. Charging Line
6. Reactor Coolant Pump Seal Water Injection and Return Lines
7. Reactor Drain Tank Discharge Line
8. Makeup Water Supply Line to the Reactor Drain Tank
9. Safety Injection Tank Fill and Drain Line

3. Fuel Handling System

a) Refueling Machine
b) Transfer Carriage System

1. Transfer Carriage System
2. Upending Machine
3. Hydraulic Power Unit

c) Fuel Transfer Tube, Valve and Flange
d) CEA Change Platform
e) Long and Short Fuel Handling Tools
f) Reactor Vessel Head Lifting Rig
g) Upper Guide Structure Lifting Rig
h) Core Barrel Lifting Rig
i) Spent Fuel Handling Machine
j) New Fuel Elevator
k) Underwater Television
l) Dry Sipping Equipment
m) Refueling Pool Seal
n) In-Core Instrumentation and CEA Cutter
o) Extension Shaft Uncoupling Tool

4. Chemical and Volume Control System

a) Pumps

1. Charging Pumps
2. Boric Acid Makeup Pumps
3. Reactor Makeup Water Pumps
4. Holdup Pumps
5. Reactor Drain Pumps

b) Tanks

1. Volume Control Tank
2. Boric Acid Batching Tank
3. Refueling Water Tank
4. Holdup Tank
5. Reactor Makeup Water Tank
6. Reactor Drain Tank
7. Equipment Drain Tank

Table 1.10-1 (Continued)

c) Heat Exchangers

1. Regenerative Heat Exchanger
2. Letdown Heat Exchanger
3. RCP Seal Injection Heat Exchanger

d) Ion Exchangers

1. Purification Ion Exchangers
2. Deborating Ion Exchanger
3. Preholdup Ion Exchanger
4. Boric Acid Condensate Ion Exchanger

e) Filters

1. Purification Filters
2. Boric Acid Filter
3. Reactor Makeup Water Filter
4. Reactor Drain Filter
5. Seal Injection Filters

f) Gas Stripper Package
g) Boric Acid Concentrator Package
h) Process Radiation Monitor
i) Boronmeter
j) Gas Stripper Effluent Radiation Monitor
k) Instrumentation and Component Controls
l) Valves
m) Chemical Addition Package

5. Shutdown Cooling System

a) Shutdown Cooling Heat Exchangers
b) Instrumentation and Component Controls
c) Valves

6. Test Programs

a) Preoperational Tests for CESSAR Design Scope Systems
b) Startup Tests for CESSAR Design Scope Systems

7. Reactor Protection System

a) Variable Overpower Trip
b) High Logarithmic Power Trip
c) High Pressurizer Pressure Trip
d) Low Pressurizer Pressure Trip
e) Low Steam Generator Pressure Trip
f) Low Steam Generator Water Level Trip
g) High Steam Generator Water Level Trip

Table 1.10-1 (Continued)

h) High Containment Pressure Trip
i) Low DNBR Trip in DNBR/LPD Calculator System
j) High Local Power Density Trip in DNBR/LPD Calculator System
k) Manual Trip

8. Supplementary Protection System

a) High Pressurizer Pressure Trip

9. Engineered Safety Features Actuation System

a) Containment Isolation Actuation Signal
b) Emergency Feedwater Actuation Signal
c) Main Steam Isolation Signal
d) Safety Injection Actuation Signal
e) Recirculation Actuation Signal
f) Containment Spray Actuation Signal

10. Control Systems

a) Reactor Regulating System
b) Control Element Drive Mechanism Control System
c) Pressurizer Pressure Control System
d) Pressurizer Level Control System
e) Feedwater Control System
f) Reactor Power Cutback System
g) Steam Bypass Control System
h) Boron Control System

11. Monitoring Systems

a) Plant Monitoring System
b) Core Operating Limit Supervisory System
c) In-core Instrumentation System

1. Fixed In-core Instrument System
2. Movable In-core Instrument System

12. Nuclear Instrumentation

a) Source Range Channels
b) Power Range Channels - Control
c) Logarithmic and Linear Safety Channels

13. Other Protective Instrumentation

a) Shutdown Cooling System Suction Line Isolation Valve Interlocks
b) Safety Injection Tank Isolation Valve Interlocks

Table 1.10-2 CESSAR Interface Summary

SER Section	Interface
3.2.1	Classification of BOP interconnections to CESSAR
3.3	Protection from wind and tornado loadings
3.4	Protection from internal and external flooding
3.5	Protection from missiles, including separation and barriers
3.6.1	Identification and protection from high and moderate energy pipe breaks
3.6.2	Pipe break parameters and restraint design
3.7	NSSS seismic design analysis and BOP structural response
3.9.2	Vibration testing program and acceptance criteria
3.9.2	Plant-specific asymmetric LOCA loads evaluation
3.9.3	Design analysis responsibilities and stress limits
3.9.3	Design and installation of pressure-relief devices
3.9.3	Support designs and mechanical design requirements
3.9.6	Inservice testing of pumps and valves
3.10	Seismic and dynamic qualification implementation
3.11	Environmental qualification implementation
4.2.1	Control rod surveillance program
4.2.3	Fuel assembly LOCA and seismic analysis
4.2.4	General fuel surveillance program
4.4.4	Loose parts monitoring program
4.4.5	CPC software implementation
5.1	Reactor coolant system design requirements
5.2.1	ASME Code Edition, Addenda, and Code cases
5.2.2	Overpressure protection design requirements

Table 1.10-2 (Continued)

SER Section	Interface
5.2.3	Reactor Coolant Pressure Boundary materials
5.2.4	RCPB inservice inspection program
5.2.5	Reactor coolant leakage detection
5.3.1	Fracture toughness for RCPB components
5.4.1.1	RCP flywheel inspection and fracture toughness data
5.4.2.2	Steam generator inservice inspection program
5.4.3	Shutdown Cooling System design requirements
5.4.4	Pressurizer Relief Tank design requirements
6.1.1	ESF metallic materials
6.1.2	Application of organic materials
6.2.1	Containment - mass and energy release data
6.2.1.2	Main steam and main feed isolation closure time
6.2.1.3	Subcompartments - mass and energy release data
6.2.1.4	Containment design parameters and impact of purging on minimum containment pressure for ECCS
6.2.2	Containment spray system design criteria
6.2.4	Containment isolation valve design requirements
6.3.2	ECCS design requirements
6.5	Postaccident sump chemistry
7.1	Instrumentation and control design requirements
7.2.4	RTSS testing
7.2.5	RPS response time testing
7.2.6	RPS setpoint implementation
7.2.7	RPS channel bypass period
7.3	ESFAS design requirements

Table 1.10-2 (Continued)

SER Section	Interface
7.3.1	ESFAS testing
7.3.4	ESFAS status requirements
7.3.5	ESFAS channel bypass period
7.3.6	ESFAS setpoint implementation
7.4.1	SDCS control and valve indication
7.4.3	Remote shutdown capability
7.4.5	Instruments and controls for safe shutdown
7.5	Safety-related display instrumentation
7.5.2	Postaccident monitoring
7.5.3	Automatic bypass indication
7.6.1	LTOP alarms
7.7.12	Control system failures
8.	Onsite and offsite power system requirements
9.1.1	New fuel storage requirements
9.1.2	Spent fuel storage requirements
9.1.4	Operation of fuel handling equipment
9.2.2	Auxiliary cooling water requirements
9.2.5	Ultimate heat sink (see 9.2.2)
9.2.6	Condensate storage design requirements
9.3.1	Instrument air design requirements
9.3.3	Process sampling requirements
9.3.4	Chemical and Volume Control System design requirements
9.4	HVAC design requirements
9.5	Fire protection

Table 1.10-2 (Continued)

SER Section	Interface
10.2.1	Main steam design requirements
10.2.2	Turbine bypass control
10.3.1	Secondary water chemistry control program
10.4	Main feedwater design and waterhammer mitigation
10.5	Auxiliary feedwater design requirements
11.1	CVCS and BRS sources
14.0	Preoperational and initial test program
15.3.9	ATWS requirements
15.4	Atmospheric dispersion, primary coolant activity, primary to secondary leakage, and containment leak rate
22.2	Plant-specific Action Plan requirements.

2 SITE CHARACTERISTICS

The detailed site characteristics will be reviewed in conjunction with applications referencing CESSAR. Appendix O to 10 CFR 50 requires that standard design applications include those site parameters postulated for the design and analysis of the standard design. CESSAR identifies site-related interface information which will assure the applicability of the CESSAR design bases. This information includes: (1) the seismic response analysis of the NSSS, (2) protection of CESSAR from site-related hazards (e.g., winds, floods and missiles), and (3) environmental conditions assumed for the accident and transient analyses.

The evaluation of these site-related interface requirements is discussed in detail in the applicable subsections of this report. In general, we conclude that the site-related interfaces provided in CESSAR adequately describe the site parameters postulated for the design, and the design has been adequately analyzed and evaluated in terms of such parameters, in accordance with Appendix O to 10 CFR 50, and are therefore acceptable.

3 DESIGN CRITERIA FOR SYSTEMS, COMPONENTS, AND EQUIPMENT

3.1 General

3.1.1 Conformance With General Design Criteria

The CESSAR FSAR describes the manner by which the CESSAR System 80 design conforms to the General Design Criteria (GDC) contained in Appendix A to 10 CFR Part 50, for the scope of the CESSAR design. The staff has reviewed the CESSAR design and concludes, subject to the additional requirements imposed by the staff and the exceptions granted as discussed in this report, that CESSAR has been designed to meet the requirements of the GDC.

3.1.2 Conformance With Industry Codes and Standards

Our review of the CESSAR systems and components relies extensively on the application of industry codes and standards that have been used as accepted industry practice. These codes and standards, as cited in this report and attached bibliography (Appendix B), have been previously reviewed and found acceptable by us; and have been incorporated into our Standard Review Plan (NUREG 0800).

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

GDC 2, "Design Bases for Protection Against Natural Phenomena," of 10 CFR Part 50, Appendix A, in part, requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety function. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to 10 CFR Part 100 guideline exposures. The earthquake for which these plant features are designed is defined as the safe shutdown earthquake (SSE) in 10 CFR Part 100, Appendix A. The SSE is based upon an evaluation of the maximum earthquake potential and is that earthquake which produces the maximum vibratory ground motion for which structures, systems, and components important to safety are designed to remain functional. Those plant features that are designed to remain functional if an SSE occurs are designated seismic Category I in Regulatory Guide 1.29. Regulatory Guide 1.29, "Seismic Design Classification," is the principal document used in our review for identifying those plant features important to safety which, as a minimum, should be designed to seismic Category I requirements. Our review of the seismic classification of systems and components (excluding electrical features) within the CESSAR scope performed in accordance with the guidance in Standard Review Plan 3.2.1, "Seismic Classification."

The systems and components important to safety that are required to be designed to withstand the effects of an SSE and remain functional have been identified in an acceptable manner in Table 3.2-1 of the CESSAR. The safety-related systems and components that are within the scope of the CESSAR design are: (1) reactor coolant system, (2) safety injection system, (3) chemical and volume control system, and (4) fuel handling system. All other structures and balance of plant systems outside the scope of the CESSAR design that are required to withstand an SSE and remain functional must be identified in the Safety Analysis Report that references CESSAR. Table 3.2-1, in part, identifies major components in fluid systems and mechanical systems designated as seismic Category I. In addition, piping and instrumentation diagrams in the CESSAR FSAR identify the interconnecting piping and valves and the boundary limits of each system classified as seismic Category I. Portions of the CESSAR systems identified above which are not required to perform a safety function such as vent lines, fill lines, drain lines and test lines on the downstream side of isolation valves are not designed to seismic Category I requirements. We have reviewed Table 3.2-1 and the fluid system piping and instrumentation diagrams, and we conclude that the systems and components important to safety that are within the scope of the CESSAR have been properly classified as seismic Category I items in conformance with Regulatory Guide 1.29, Revision 1, except for those items discussed below.

We have identified two items which require clarification or are exceptions to Regulatory Guide 1.29. They are:

(1) Regulatory Position C.1.b of the guide identifies "the reactor core and reactor vessel internals" as seismic Category I. The application of this seismic Category I classification in the CESSAR is limited to "those core support structures necessary to support and restrain the core and to maintain safe shutdown capability." Failure of other reactor internals that are not designed to seismic Category I requirements will have no impact on the reactor core and reactor vessel internals important to safety (seismic Category I items) since these components are designed and constructed such that their failure would not adversely affect the functioning of the seismic Category I components within the reactor vessel. We find this position to be acceptable.

(2) Regulatory Position C.1.h of the guide identifies cooling water and seal water systems to the reactor coolant pumps as seismic Category I. In CESSAR, the cooling water supply to the pump bearing motor, pump seal coolers, and pump motor coolers are classified as nonseismic Category I.

Testing for the loss of component cooling water to a CE-KSB production reactor coolant pump has been completed and is documented in CENDP-201-A Supplement 1. The results of those tests show that there is sufficient margin in the design of the reactor coolant pumps so that they will be able to perform their intended safety function after an SSE without component cooling water. CENPD-201-A Supplement 1 has been reviewed by the staff (see Section 9.2.2 of this Safety Evaluation Report) and approved as an acceptable reference for this exception to Regulatory Guide 1.29. Consequently, the cooling water supply to the reactor coolant pumps within the BOP scope does not necessarily have to be designed seismic Category I.

CESSAR defines the interfaces between the BOP structures, systems, components, and supports and those systems and components within the CESSAR scope. Our evaluation included an evaluation of the interfaces defined by CE to assure compatibility of the seismic design and classification across these interfaces.

Based on our evaluation, we conclude that the interface requirements in Section 3.2.1 of CESSAR are acceptable and complete. We conclude that the systems and components important to safety that are within the scope of the CESSAR are designed to withstand the effects of an SSE and remain functional and are properly classified as seismic Category I items in accordance with Regulatory Guide 1.29, or are acceptable exceptions and clarifications as noted above, and constitute an acceptable basis for satisfying the applicable portions of the requirements of GDC 2, and are, therefore, acceptable.

3.2.2 System Quality Group Classification

GDC 1, "Quality Standards and Records," of 10 CFR Part 50, Appendix A requires that nuclear power plant systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. These fluid system pressure-retaining components are part of the reactor coolant pressure boundary and other fluid systems important to safety, where reliance is placed on these systems: (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintain it in a safe shutdown condition, and (3) to retain radioactive material. Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," is the principal document used in our review for identifying on a functional basis the components of those systems important to safety that are Quality Groups A, B, C, and D. Section 50.55a of 10 CFR Part 50 identifies those American Society of Mechanical Engineers (ASME) Section III, Class 1 components that are part of the reactor coolant pressure boundary (RCPB). Conformance of these RCPB components with Section 50.55a of 10 CFR 50 is discussed in Section 5.2.1.1 of this report. These RCPB components are designated in Regulatory Guide 1.26 as Quality Group A. Certain other RCPB components which meet the exclusion requirements of footnote 2 of the rule are classified Quality Group B in accordance with Regulatory Guide 1.26. Our review of the quality group classification of pressure-retaining components of fluid systems important to safety of the CESSAR was performed in accordance with the guidance in Standard Review Plan 3.2.2, "System Quality Group Classification."

The systems and components important to safety have been identified in an acceptable manner in Table 3.2-1 of the CESSAR. These systems and components that are within the scope of the CESSAR design are: (1) reactor coolant system, (2) safety injection system, (3) chemical and volume control system, and (4) fuel handling system. All other balance of plant systems important to safety that are outside the scope of the CESSAR design must be identified in the Safety Analysis Report that references CESSAR. Table 3.2-1, in part, identifies the major components in fluid systems such as, pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves and mechanical systems, such as the spent fuel handling machine. The piping and instrumentation diagrams in CESSAR identify the classification boundaries of the interconnecting piping and valves. In addition, the classification requirements at the interface of systems and components within the scope of the CESSAR and with the balance of plant systems and components are properly identified. We have reviewed Table 3.2-1 and the fluid system piping and instrumentation diagrams and we conclude that pressure-retaining components have been properly classified in conformance with Regulatory Guide 1.26, except for the following item.

Regulatory Position C.2.b of the guide identifies cooling water and seal water systems to the reactor coolant pumps as constructed to Quality Group C standards. As noted in Section 3.2.1 of this Safety Evaluation Report, testing for the loss of component cooling water to the reactor coolant pumps has been completed. This testing has been documented in CENPD-201-A Supplement 1 which has been reviewed by the staff (see SER Section 9.2.2) and approved as an acceptable reference for this exception to Regulatory Guide 1.26. Consequently, the cooling water systems to the reactor coolant pump within the BOP scope may be acceptably classified as Quality Group D.

The codes and standards used in the construction of components are identified in Table 3.2-2 of the CESSAR. We find this summary list of codes and standards used in the construction of components to be acceptable. CE has utilized the American Nuclear Society (ANS) Safety Classes 1, 2, 3 and Non-Nuclear Safety (4) as defined in ANSI-18.2a-1975, "Revision and Addendum to Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," in the classification of system components. Safety Classes 1, 2, 3 and 4 correspond to the NRC Quality Groups A, B, C, and D in Regulatory Guide 1.26 and have been used by the CE as an alternate to the NRC quality group designations.

A summary of the relationship of the NRC Quality Group and ANS Safety Classes are as follows:

NRC Quality Group	CE PWR Safety Class
A	1
B	2
C	3
D	Non-nuclear Safety (4)

We have reviewed the use of ANS Safety Classes in Table 3.2-1 and we find the classification of components to be acceptable. Quality Group A (Safety Class 1) components of the RCPB are constructed[1] in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Class 1. Components in fluid systems important to safety that are classified Quality Group B (Safety Class 2) are constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Class 2. Components in fluid systems important to safety that are classified Quality Group C (Safety Class 3) are constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Class 3. Components in fluid systems that are classified Quality Group D (Safety Class 4) are constructed to the following codes as appropriate: ASME Boiler and Pressure Vessel Code, Section VIII, Division 1 or 2; ANSI B31.1.0 Power Piping; and storage tank codes, such as API-620 and API-650.

We conclude that construction of the components in fluid systems important to safety in conformance with the ASME Code, the Commission's regulations, and the guidance provided in Regulatory Guide 1.26 and ANSI-N18.2a, provides assurance that component quality is commensurate with the importance of the safety function of these systems and constitutes an acceptable basis for satisfying the requirements of GDC 1 and is, therefore, acceptable.

[1]Constructed, as used herein, is an all-inclusive term comprising materials certification, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components.

3.3 Wind and Tornado Loadings

The design criteria for wind and tornado loadings are outside the scope of CESSAR and will be evaluated in the applications referencing CESSAR. However, the CESSAR FSAR includes an interface requirement for each safety-related system in Sections 3.3, 5.4.7, 5.1.4, 6.3.1, 6A-7, 6B-7, 7.1.3, and 9.3.4 which requires that protection be provided for wind and tornado loadings. This interface conforms to the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and is, therefore, acceptable and complete.

3.4 Water Level (Flood) Design

CESSAR indicates that protection for safety-related equipment from the effects of flooding is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements, in Sections 3.4, 5.4.7, 5.1.4, 6.3.1, 6A-7, 6B-7, 7.1.3, and 9.3.4, concerning location and installation of protection for safety-related systems and components within the CESSAR scope from both internal flooding and external flooding (the probable maximum flood) in order to assure their safety function in accordance with the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."

Based on our review of the CESSAR interfaces, we conclude that CESSAR provides adequate information concerning flood potection for essential systems in order that referencing applicants can comply with the requirements of GDC 2 and the guidelines of Regulatory Guide 1.102 and is, therefore, acceptable and complete in this regard.

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

CESSAR indicates that protection of safety-related systems inside and outside containment from the effects of internally generated missiles is the responsibility of applicants that reference CESSAR. However, CESSAR includes interface requirements in Sections 5.4.7, 5.1.4, 6.3.1, 6A-7, 6B-7, 7.1.3, and 9.3.4, concerning location, separation and barriers, as necessary, for protection of safety-related systems and components outside containment within the CESSAR scope from internally generated missiles in order to assure their safety function and assuring a safe plant shutdown in accordance with the requirements of GDC 4, "Environmental and Missile Design Bases." In addition, CESSAR provides an evaluation of potential missile sources from rotating and pressurized equipment outside containment in the CESSAR scope on the basis that a single failure could result in missile generation which indicates that these components are not sources of internally generated missiles based on their design. We concur with this assessment.

Based on our review of CESSAR and the CESSAR interfaces, we conclude that CESSAR provides adequate information concerning protection of essential systems from internally generated missiles outside containment in order that referencing applicants can comply with the requirements of GDC 4 and has satisfactorily evaluated potential missile sources outside containment from components within the CESSAR scope and is, therefore, acceptable and complete in this regard.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Protection against postulated internally generated missiles inside containment associated with plant operation such as missiles generated by rotating or pressurized equipment as identified in the requirements of GDC 4, "Environmental and Missile Design Bases" is provided by any one or a combination of barriers, separation, and equipment design. Protection is provided to assure against the occurrence of a loss of coolant accident due to missile impact and to maintain the capability for a safe shutdown. The primary means of protection for safety-related equipment from damage resulting from internally generated missiles is provided by the primary shield walls and separation within the containment.

CESSAR indicates that protection of safety-related systems and components inside containment from the effects of internally generated missiles is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Sections 3.5.1.2, 5.4.7, 5.1.4, and 9.3.4 concerning location, separation, and barriers as necessary for providing protection for safety-related systems and components inside containment within the CESSAR scope from internally generated missiles in order to assure their safety function and assuring a safe plant shutdown in accordance with the requirements of GDC 4.

CESSAR has provided an evaluation of potential missile sources inside containment from systems and components within the CE scope. The only credible potential missile sources identified are from high energy systems on the basis that a single failure generates the missile. These potential missile sources are:

(1) Reactor vessel

- (a) closure head nut
- (b) closure head nut and stud
- (c) control rod drive assembly

(2) Steam generator

- (a) primary manway stud and nut
- (b) secondary handhole stud and nut
- (c) secondary manway study

(3) Pressurizer

- (a) safety valve with flange
- (b) safety valve flange bolt
- (c) lower temperature element
- (d) manway stud and nut

(4) Main coolant piping temperature nozzle with resistance temperature detector

(5) Surge and spray piping thermowells with resistance temperature detector

(6) Reactor coolant pump thermowell with resistance temperature detector

Kinetic energy and impact cross section were determined for each of the above potential missiles. CESSAR indicates that it is the responsibility of applicants referencing CESSAR to verify that adequate structures, shields, or separation barriers are provided for protection of safety-related equipment from the above primary missiles. We concur with the above postulated missile sources and their characteristics. We further conclude that CESSAR provides sufficient information for the applicant to provide the necessary protection from the above postulated missiles inside containment using the interface requirements previously described.

In addition, CESSAR has analyzed the potential for missile sources as a result of failures in the reactor coolant pump (RCP) flywheel in accordance with the guidelines of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity." The CESSAR analysis evaluated the materials integrity of the flywheel under assumed overspeed conditions of the pump as a result of pipe break at the pump section and discharge. This analysis verified that failure of the flywheel does not occur and thus it is not a postulated missile source. Refer to Section 5.4.1.1 of this report for further discussion on the material integrity of the RCP flywheel. We concur with the CESSAR analysis.

We have reviewed the adequacy of the CESSAR evaluation and information provided in order to maintain the capability for a safe plant shutdown and prevent the occurrence of a loss-of-coolant accident in the event of internally generated missiles inside containment. Based on the above, we conclude that CESSAR has provided on adequate analysis and complete information, regarding the missile sources in the CESSAR scope, for the referencing applicant to assure that the plant design is in conformance with the requirements of GDC 4 with respect to missile protection for internally generated missiles inside containment and is, therefore, acceptable. We further conclude that CESSAR complies with the guidelines of Regulatory Guide 1.14 concerning reactor coolant pump flywheel integrity and is, therefore, acceptable.

3.5.1.3 Missiles Generated by Natural Phenomena

CESSAR indicates that the identification and evaluation of missiles generated by natural phenomena is a site specific matter that is the responsibility of applicants that reference CESSAR. Interface requirements are described in Section 3.5.2.

3.5.2 Structures, Systems and Components to be Protected from Externally Generated Missiles

CESSAR indicates that protection for safety-related equipment from the effects of externally generated missiles (tornado missiles) is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Sections 3.5.3, 5.1.4, 5.4.7, 6.3.1, 6A-7, 6B-7, 7.1.3, and 9.3.4, concerning protection of safety-related systems and equipment within CESSAR scope from externally generated missiles in order to assure their safety function in accordance with the requirements of GDC 4, "Environmental and Missile Design Bases."

Based on our review of the CESSAR interfaces, we conclude that CESSAR provides adequate information concerning protection of essential systems from externally gnerated missiles in order that referencing applicants can comply with the requirements of GDC 4 and is, therefore, acceptable, and complete in this regard.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containments

CESSAR indicates that protection of safety-related equipment from the effects of postulated piping failures in high and moderate energy fluid systems outside containment and the specific NRC criteria to be applied in providing this protection is the responsibility of applicants that reference CESSAR. All fluid piping systems outside the containment are in the BOP scope. However, the CESSAR FSAR includes interface requirements in Sections 3.6.1, 5.1.4, 5.4.7, 6.3.1, 6A-7, 6B-7, 7.1.3, and 9.3.4 concerning identification of high and moderate energy piping systems and protection of safety-related systems and components within the CESSAR scope from the effects of failures in fluid systems, including the effects of water spray and jet impingement outside containment in order to assure their safety function in accordance with the requirements of GDC 4, "Environmental and Missile Design Bases."

Based on our review of the CESSAR interfaces, we conclude that CESSAR provides adequate information concerning protection of essential systems from the effects of high and moderate energy piping failures outside containment in order that referencing applicants can comply with the requirements of GDC 4 and is, therefore, acceptable and complete in this regard.

3.6.2 Determination of Break Locations and Dynamic Effects Associated With the Postulated Rupture of Piping

GDC 4, "Environmental and Missile Design Bases" requires that structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power plant.

Our review, conducted in accordance with Standard Review Plan Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated With the Postulated Rupture of Piping," pertains to the methodology used for protecting safety-related structures, systems and components against the effects of postulated pipe breaks both inside and outside containment. The effect that breaks in high energy fluid systems would have on adjacent safety-related structures, systems, or components has been reviewed with respect to jet impingement and pipe whip. We also reviewed the location, size, and orientation of postulated failures and the methodology used to calculate the resultant pipe whip and jet impingement loads which might affect nearby safety-related structures, systems, or components.

CE has identified CESSAR break locations and type based on the methodology of topical report CENPD-168A, "Design Basis Pipe Breaks for the CE Two Loop Reactor Coolant System." This topical report was approved by the staff on May 5, 1977 for use on previous Combustion Engineering two-loop plants. The results of the CESSAR analysis are summarized in the FSAR. Terminal ends are selected as break locations regardless of the stress intensity or cumulative usage factor. The range of primary plus secondary stress intensity and the cumulative usage factor is shown for all intermediate break locations.

The break locations identified in CESSAR analysis are identical with those reported in CENPD-168A. We require that a plant-specific analyses be done for each plant referencing CESSAR to confirm that the postulated break locations are within the design envelope of CESSAR, as described below.

The pipe whip dynamic analysis methods of CENPD-168A are applied to the CESSAR reactor coolant system main loop piping to establish pipe whip restraint design requirements. The specific design of pipe whip restraints for the CESSAR reactor coolant system is not within the CESSAR scope. CESSAR provides a range of parameters required for the design of pipe whip restraints to any applicant utilizing the CE System 80 reactor coolant system. The range of parameters provided by CESSAR limit the area of guideline breaks in the CE System 80 reactor coolant piping to those given in Table 3.6-2 of CESSAR.

We have reviewed the procedures used to determine the type of break, break location, and break area used in the design of the CE System 80 reactor coolant system and find them to be acceptable. We will require that the following additional information be included in each Safety Analysis Report referencing CESSAR:

(1) Actual limited break flow areas and the separation time used for any circumferential break location in the plant-specific reactor coolant loop.

(2) Assurance that the system parameters (e.g., operational transients) used for the plant-specific analyses fall within the design envelope of CESSAR.

(3) Parameters used in the design of the pipe whip restraints for the plant-specific reactor coolant loop such as restraint stiffness, initial gap size, and type and location of attachments to piping which limit pipe motion due to a postulated break.

(4) The forcing functions used in the analysis of the plant-specific reactor coolant system.

(5) A plant specific analysis to confirm that postulated break locations are within the CESSAR design envelope.

Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," provides criteria for determining high and moderate energy lines. The criteria for determining high and moderate energy lines in CESSAR conforms to Regulatory Guide 1.46 and are, therefore, acceptable. In addition, CESSAR specifies that a list of all high and moderate energy lines important to safety will be included in each Safety Analysis Report referencing CESSAR. We will review the list of high and moderate energy lines on each reference plant application to assure proper implementation of the criteria above.

The effects of jet impingement, caused by postulated piping breaks within the CESSAR design scope, on structures, systems, or components outside the CESSAR design scope have been reviewed. CE provides jet impingement loads and mass energy releases for specific break locations within the CESSAR design scope to the applicant referencing CESSAR so that the effects on BOP structures, systems, or components can be evaluated. We have reviewed the methods described in CESSAR for determining (1) jet impingement loads and (2) break areas and break separation times for calculating mass and energy releases. We conclude that these methods conform to the guidelines of Branch Technical Position MEB 3-1 and the Standard Review Plan Section 3.6.2 and are, therefore, acceptable. We will review the application of these methods in the applications referencing CESSAR.

In addition to the total plant inspection program required by Section XI of the ASME Code, the Standard Review Plan Section 3.6.2 sets forth certain criteria for the analysis and subsequent augmented inservice inspection of high energy piping within the break exclusion area of the containment penetration region. CESSAR indicated that all details concerning inservice inspection of piping within the break exclusion region will be addressed in the applications refer-ening CESSAR.

Protective assemblies for containment penetration piping are not within the CESSAR scope of design. CESSAR indicates that details of these assemblies will be addressed in the applications referencing CESSAR.

The methods used for determining the location, type, and effects of postulated pipe breaks in the primary reactor coolant loop have been presented. The effects resulting from these postulated pipe failures will be used to evaluate the design of structures, systems, and components required to bring the plant to a safe shutdown condition or mitigate the effects of the postulated piping failures. Where these safety-related structures, systems, or components are outside of the CESSAR design scope, the information required for each plant design is provided to the reference plant by CE as an interface requirement. CESSAR further indicates that pipe whip restraints, jet impingement barriers, and other such devices will be used in reference plants to mitigate the effects of the postulated piping failures in the primary loop. By the nature of this interface, we cannot conclude on the completeness of this interface requirement. However, the methods described in CESSAR are acceptable. We will review the procedures used in the design of pipe whip and jet impingement restraints, along with the plant-specific analysis details previously described, in each application referencing CESSAR.

We have reviewed the methods in CESSAR and have concluded that they provide for a spectrum of postulated pipe breaks which includes the most likely location for piping failures in the primary coolant loop, and that the type of breaks and their effects have been conservatively estimated. We find that the information provided by CE to applicants utilizing CESSAR for the design of pipe whip restraints on the CE System 80 reactor coolant system is adequate and provides assurance that the restraints will be designed to function properly in the event of a postulated piping failure. We further conclude that the pipe failure methodology used in CESSAR for the design of structures, systems, and components necessary to safely shutdown the plant and to mitigate the consequences of these postulated piping failures provides reasonable assurance of their ability to perform their safety function following a failure in the primary reactor coolant loop. The methods used in CESSAR comply with Standard Review Plan Section 3.6.2 and satisfy the applicable portions of GDC 4 for the CE System 80 piping and, therefore, are acceptable for any plant referencing CESSAR.

3.7 Seismic Design

The seismic design of CESSAR was conceived on the basis of an envelope design of, and improved modifications to, previous CE two-loop nuclear steam supply systems. The adequacy of the final design of CESSAR will be verified for each reference plant by using plant-specific loadings. By referencing CESSAR, each applicant has committed to confirm the seismic design adequacy of the primary system to the site-specific seismic loads, such that conformance with the guidelines of Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," by the reference plant applicant, will apply also to the CESSAR scope. The verification of the results of the CESSAR plant-specific analysis, as described below, is an interface requirement for the design of balance of plant seismic Category I structures, systems, and components.

All seismic Category I systems and components in the CESSAR design scope are designed to the appropriate criteria of the ASME Code, Section III. These criteria are discussed in Section 3.9.3 of this report. The major components in the CESSAR reactor coolant loop are the reactor vessel, the steam generator, the reactor coolant pumps, the reactor coolant piping, and the pressurizer. The design adequacy of the major components in the CESSAR reactor coolant loop design is verified by dynamic analysis methods employing either time history or response spectrum techniques.

As part of the design interface between CESSAR and the BOP, dynamic coupling effects between the containment internal support structure and the reactor coolant system must be considered. A detailed mathematical model of the CESSAR reactor coolant system is provided by CE to the applicant referencing CESSAR for inclusion in the analysis of the plant-specific containment internal support structure. The results of the analysis of the containment internal support structure include several time history forcing functions. These time histories are provided by the applicant of the reference plant to CE for use in a separate, more detailed analysis of the CESSAR reactor coolant system. A similar technique is used for the analysis of BOP piping system connections to the CESSAR systems, as described in Section 3.9.3. Based on our review of the analytical methods and modeling techniques described in the CESSAR FSAR, our audit of the detailed design analyses, and the confirmatory analysis described in Section 3.9.1 of this report, we conclude that the procedures used to account for the coupling effects between the CESSAR systems and the BOP systems and structures are acceptable.

To account for possible dynamic interaction effects between the components of the reactor coolant system, a composite coupled model is used. This model includes the reactor vessel, the two steam generators, the four reactor coolant pumps, and the interconnecting reactor coolant piping. The analysis of these dynamically coupled multisupported components utilizes different time history input excitations applied simultaneously at each support location. By using this technique, the effect of differential seismic displacements on the equipment and supports is considered. Sufficient detail of the reactor internals is included in the model to assure that possible dynamic coupling from the reactor coolant system to the reactor internals is considered. The results of the analysis of the coupled components of the reactor coolant system include a time history forcing function that is used in a separate, more detailed analysis of the reactor internals.

The analysis of the reactor internals includes a linear vertical analysis. If the linear vertical analysis for a specific plant indicates that the response of the core is sufficiently large to cause it to lift off the core plate, a vertical nonlinear analysis of the internals will also be performed. CESSAR indicates that, if the vertical nonlinear analysis is performed, the results of the analysis will be included in the reference plant Safety Analysis Report.

The only seismic Category I piping included in the CESSAR scope of design is the reactor coolant loop piping. The piping associated with the CESSAR reactor coolant loop is included in the composite coupled model of the reactor coolant system. The safety injection and pressurizer surge lines are decoupled from the primary coolant loop and are within the design scope of the applicant referencing CESSAR. CE provides the applicant of each plant referencing CESSAR with the locations of all balance-of-plant piping interfaces with the CESSAR components. CE specifies allowable design limits at each interface location to assure that the components within the CESSAR design scope can perform their intended functions. Based on our evaluation of the analytical methods described in CESSAR and our audit of the interface information provided to the applicant of each plant referencing CESSAR, we conclude that the procedures used for assuring the seismic adequacy of all seismic Category I piping in the CESSAR scope of design are acceptable.

Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," contains recommended values of damping for the seismic analysis of structures, systems, and components. The damping values used in CESSAR are the same as those specified in Regulatory Guide 1.61 and are acceptable.

Three spatial components of earthquake motion are considered in the seismic analysis of all seismic Category I systems and components in the CESSAR scope of design. The three spatial components of earthquake motion are combined by the square-root-of-the-sum-of-the-squares (SRSS) method. Modal responses are combined using SRSS when the modal response spectrum method of analysis is used. However, in the analysis of simple systems where three or less dynamic degrees of freedom are involved, the modal responses are combined by the absolute sum method. In addition, closely spaced modes, those having frequencies that lie within 10% of each other, are combined by the absolute sum method before being combined with the other significant modes by SRSS. We find that this is an acceptable method for combining seismic response and that it complies with the requirements of Regulatory Guide 1.92, "Combining Model Responses and Spatial Components in Seismic Response Analysis."

A commitment is provided in CESSAR that the seismic adequacy of the major components in the CESSAR reactor coolant system will be verified using the plant specific seismic loading and that the results will be included in any Safety Analysis Report that references CESSAR and will verify that the CESSAR components meet the load combinations and stress limits in Section 3.9.3 of the CESSAR FSAR. We find this to be acceptable.

The scope of the review of the seismic system analysis for the CESSAR scope of design included the seismic analysis methods for the primary reactor coolant system and the commitment to perform plant-specific analyses. We have reviewed the techniques used for modeling and evaluating seismic Category I systems and components within the CESSAR design. The review also included the criteria used and the seismic analysis techniques utilized for the reactor internals. There is no interaction between nonseismic Category I and seismic Category I piping in the CESSAR scope of design.

The dynamic system analysis for the CESSAR System 80 reactor coolant system complies with Standard Review Plan Section 3.7.3 and Regulatory Guides 1.61 and 1.92 and constitute an acceptable basis for satisfying the applicable portions of General Design Criterion 2 and, therefore, is acceptable.

3.8 Design of Category I Structures

The design of structures is outside the scope of CESSAR and will be evaluated in the applications referencing CESSAR. However, CESSAR does provide an interface, as described in Section 3.7, to assure that the design of structures is compatible with the design of systems within the CESSAR scope.

3.9 Mechanical Systems and Components

The review performed under Standard Review Plan Sections 3.9.1 thro 3.9.6 pertains to the structural integrity of various safety-related mechanical components and supports in the CESSAR scope of design.

Our review was not limited to those components covered by the ASME Code, but was extended to other components such as control element drive mechanisms and certain reactor internals designed to industry standards other than the ASME Code. We reviewed such issues as load combination, allowable stresses, methods of combination and analysis, summary of results, requirements for preoperational testing, and requirements for inservice testing of pumps and valves. Our review concludes that there is adequate assurance of the mechanical components performing their safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

The review of this section of CESSAR was performed following Standard Review Plan 3.9.1, "Special Topics for Mechanical Components." We have reviewed the design transients and methods of analysis used for all seismic Category I core support structures, components, and their supports including those designated as Class 1, 2, 3 or CS under the ASME Code, Section III, and those not covered by the Code. The assumptions and procedures used for the inclusion of transients in the fatigue analysis of ASME Code Class 1 and CS components have also been reviewed. Our review also covered the computer programs used in the design of seismic Category I mechanical components in the CESSAR scope of design. Experimental and inelastic analytical techniques have been covered by our review.

Additionally, we have contracted with the Pacific Northwest Laboratory to perform an independent analysis of the reactor coolant piping system within the CESSAR scope of design. This analysis verified that the CESSAR piping system met the applicable ASME Code acceptance requirements. The detailed results of this analysis are documented in the report by Pacific Northwest Laboratory, "CE System 80 -Loop 1 Piping Analysis," dated September 1981.

The design transients used for the evaluation of transient responses and the fatigue analysis of components within the CESSAR design scope are given in Table 3.9.1-1 of CESSAR. In addition, CE provides a detailed list of transients to each applicant utilizing the CESSAR reactor coolant system. The CE supplied transients are used by the applicant in the fatigue analysis of components outside of the CESSAR scope of design, that are influenced by transients originating in the CESSAR reactor coolant system.

Based on our review of the list of design transients given in Section 3.9.1 of the CESSAR FSAR and on our audit of the list of transients in the interface documentation, we find that the design transients used in the CESSAR scope of design and the procedures for ensuring that their effects are considered in the design of systems and components in the BOP scope of design is acceptable and the list of design transients is complete.

We have reviewed the computer programs used in the analysis of the mechanical components in the CESSAR design scope. A list of the computer programs used in the static and dynamic analyses to determine the structural integrity and functional capability of these components is included in the CESSAR FSAR along with a brief description of each program. Design control measures, which are required by 10 CFR Part 50, Appendix B, require that methods of verification for all computer programs used in design be provided. Methods of verification for all computer programs have been included in the CESSAR FSAR.

Experimental stress analysis has not been used on any component in the CESSAR scope of design.

Inelastic methods of analysis have been used in the CESSAR scope of design for the design of load limiting devices on the reactor vessel lower key horizontal supports. The load limiting devices are designed to remain elastic for all normal, upset, and SSE loading conditions. For loads resulting from postulated pipe breaks, the load limiting devices are designed to deflect plastically and a nonlinear analysis is used to calculate the distribution of the loads on the system supports. The load limiting devices will be tested to demonstrate that the deformation characteristics are acceptable. The test results will be provided to the staff upon completion of the test.

The criteria used in defining the applicable transients, the computer programs used in the analysis, and the analytical methods used in design provides assurances that the stresses, strains, and displacements calculated for the mechanical components within the CESSAR design scope are as accurate as the current state-of-the-art permits and are adequate for the design of these items.

The methods of analysis that are used in CESSAR for the design of seismic Category I ASME Code Class 1, 2, 3 and CS components, component supports, reactor internals and other non-Code items are in conformance with Standard Review Plan 3.9.1 and satisfy the applicable portions of GDC 2, 4, 14 and 15 and, therefore, are acceptable.

3.9.2 Dynamic Testing and Analysis of System, Components, and Equipment

We have reviewed the methodologies, testing procedures, and dynamic analyses employed, for the CESSAR scope of design, to ensure the structural integrity and functionality of piping systems, mechanical equipment, reactor internals, and their supports under dynamic and vibratory loadings. The principal document used in our review was Standard Review Plan Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment". This review covers several areas, each of which is described below.

Piping vibration, thermal expansion, and dynamic effects testing are to be conducted during preoperational startup testing for each plant utilizing the CESSAR design. Systems to be monitored, within the CESSAR scope of design, include all ASME Code Class 1 and 2 systems. The test program must comply with the ASME Code, Section III, paragraphs NB-3622.3 and NC-3622.3 which also requires that the applicant be responsible, by observation during startup or initial operations, for ensuring that the vibration of piping systems during plant operation is within acceptable levels.

Piping system vibration may be caused by plant transients or by steady-state vibration associated with normal plant operation. This steady-state vibration, whether flow-induced or caused by nearby vibrating machinery, may cause 10^8 or 10^9 cycles of stress in the pipe during its 40 year life. For this reason, we require that the stress associated with the pipe deflection due to steady-state vibration be limited to 50% of the alternating stress intensity, Sa, at 10^6 cycles, as defined in the ASME Code, Appendix I, Figure I-9.0. We require that any application referencing CESSAR contain the details of the preoperational and startup test program, including acceptance limits for transient and steady-state vibration of the CESSAR coolant loop.

A preoperational testing of the reactor internals for flow-induced vibrations will be performed. The purpose of the preoperational vibration test program is to verify the design adequacy of the reactor internals under loading conditions that will be comparable to those experienced during plant operation.

Palo Verde Unit 1 will be the prototype plant for the CESSAR reactor internals. Preoperational and startup testing will be performed on the reactor internals in accordance with Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Startup Testing," Revision 2 for prototype plants. The Precritical Vibration Monitoring Program (PVMP) has been developed for the CESSAR reactor internals. The CESSAR program utilizes the experience from earlier CE plants. The CESSAR program includes predictions, measurements and evaluations of the core support barrel, lower support structure and upper guide structure assemblies. The program consists of four phases: (1) vibration analysis phase, (2) vibration measurement phase, (3) inspection phase, and (4) evaluation and documentation phase. CE has committed to provide the staff details of the testing program in accordance with the schedule delineated in Regulatory Guide 1.20. A final report will be issued to summarize the results of the PVMP program. We will require that CE provide a summary of the test program upon its completion.

The reactor coolant system components could be subjected to dynamic asymmetric loadings resulting from a postulated double-ended rupture of the primary coolant system piping. As a result of a generic investigation in response to

Task Action Plan A-2, the staff has developed guidelines and criteria for the evaluation of these asymmetric loads. These guidelines and criteria are presented in NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary System," dated January 1981.

We have reviewed CESSAR for the evaluation of NSSS components to the faulted loads. However, only certain aspects of the A-2 asymmetric loads methodology development are applicable to CESSAR. These are:

(1) definition of pipe breaks,

(2) methodology for the calculation of mass and energy releases,

(3) methodology for the calculation of blowdown loads, and

(4) the modelling of the vessel internals and fuel.

The above items are presented in CESSAR. We have reviewed the above items and we conclude that the methodologies presented in CESSAR conform to the guidelines in NUREG-0609 and are acceptable.

The asymmetric loads evaluation is plant specific. Each plant referencing CESSAR is required to address the following items relative to the asymmetric loads issue:

(1) definition of plant geometries,

(2) calculation of subcompartment pressurization,

(3) evaluation of building walls and foundations,

(4) structural analysis of reactor coolant system,

(5) evaluation of vessel, RCS, and ECCS piping support loads,

(6) structural analysis and evaluation of internals and CEDMs,

(7) analysis and evaluation of ECCS piping attached to RCS,

(8) structural analysis and evaluation of fuel, and

(9) overall summary of results for each plant.

Based on our review of CESSAR Section 3.9.2 our findings are as follows:

The vibration, thermal expansion, and dynamic effects test program to be conducted on systems and components in the CESSAR scope of design will be included in applications that references CESSAR. Palo Verde Unit 1 will be the prototype, in accordance with Regulatory Guide 1.20, for the reactor internals. CE has committed to a reactor internals testing program and schedule consistent with the requirements of Regulatory Guide 1.20. However, CE has not yet provided a detailed description of this program. We will report on the resolution of this issue in a revision to this report.

The dynamic system analysis performed on the systems and components in the CESSAR scope of design provide an acceptable basis for confirming the structural design adequacy of the reactor internals and the unbroken piping loops to withstand the combined dynamic loads due to a simultaneous loss-of-coolant accident and the safe shutdown earthquake. The analysis also takes into account asymmetric and subcompartment pressurization loads. These analyses provide adequate assurances that the combined stresses and strains in the components of the reactor coolant system and reactor internals do not exceed the allowable stress and strain limits for the materials of construction, and that the resulting deflections or displacements at any structural element of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The methods used for components analysis have been found to be compatible with those used for systems analysis. The proposed combinations of component and system analyses are, therefore, acceptable. The assurance of structural integrity of the reactor internals under loss-of-coolant accident and safe shutdown earthquake conditions for the most adverse postulated loading event provides added confidence that the design will withstand a spectrum of lesser pipe breaks and seismic loading events. Accomplishment of the dynamic system analysis constitutes an acceptable basis for satisfying the applicable requirements of GDC 2 and 4 and, therefore, is acceptable.

3.9.3 ASME Code Class 1, 2, and 3 Components, Components Supports and Core Support Structures

Our review under Standard Review Plan 3.9.3 is concerned with the structural integrity and functionality of pressure-retaining components, their supports, and core support structures which are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. This review is divided into several parts, each of which is discussed below.

The first area of our review covered the loading combination and the allowable stresses used in the analysis of systems and components in the CESSAR scope of design. ASME Section III, Code Class 1, components in the CESSAR licensing scope are limited to the reactor coolant system main coolant loop including the pressurizer. Loading combinations for ASME Code Class 1 components are provided in Table 3.9.3-1 of CESSAR. System responses due to a simultaneous loss-of-coolant accident (LOCA) and safe shutdown earthquake (SSE) are combined by the square-root-of-the-sum-of-the-squares, or, in some cases, by the absolute sum method.

Stress limits for ASME Code Class 1 components and piping is provided in Table 3.9.3-1 of CESSAR. Stress limits for Class 2 and 3 components are as given in Section 3.9.3.1.3 of CESSAR. We have reviewed the stress limits for ASME Code Class 1, 2, and 3 components and find them acceptable.

CE has responsed, in a letter from A. E. Scherer to C. Grimes dated October 14, 1981, to our concern that essential piping within the CESSAR design scope maintain its functional capability for all designated loading combinations evaluated to Service Levels C and D. The only piping system within the CESSAR design scope is the ASME Class 1 reactor coolant system (RCS) main loop piping. CE has performed a finite element analysis of the RCS suction leg elbow using a bending moment in excess of the maximum allowable Service Level D moment (56.7 x 10^6 in-lb). A stress-strain curve for SA 516 Gr 70 piping material at 650°F was obtained by conventional testing at the CE Metallurgy Laboratory and was used as input in the MARC computer program for the plasticity analysis. The maximum pipe deformation calculated from the analysis resulted in a strain of 0.007 in/in. The piping deformation was shown to be acceptable to demonstrate the functional capability of the RCS piping.

In order to assure that the systems and components within the CESSAR design scope can perform their intended safety functions, CE provides detailed interface documents to each plant utilizing the CESSAR design. These interface documents describe the division of responsibilities between CE and the architect-engineer and provide the architect-engineer with the necessary design requirements to assure that the CESSAR design scope systems and components can fulfill their safety functions. The scope of our review included an evaluation of the mechanical design interfaces defined in CESSAR in accordance with the guidelines contained in Regulatory Guide 1.70, Appendix A, and the criteria in the Standard Review Plan, Section 3.9.3. Based on our review, we conclude that there is reasonable assurance that the interface information provided by CE, in conjunction with the quality assurance programs, will assure an acceptable design basis for the BOP. However, we will require that each reference plant application identify any unique design features which might require special interface design considerations. We will review these design features to assure that they are compatible with the CESSAR design.

We find that the specified design and service combinations of loadings as applied to ASME Code Class 1, 2, and 3 pressure retaining components in systems designed to seismic Category I standards are such as to provide assurance that, in the event of an earthquake affecting the site or other service loadings due to postulated events or system operating transients, the resulting combined stress imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provide a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity. The design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 components constitute an acceptable basis for design and satisfy the applicable portions of GDC 1, 2, and 4.

An interface requirement is provided in CESSAR that details of the design and installation of pressure relieving devices will be included in each Safety Analysis Report that references CESSAR.

The final area of our review in this section is the criteria used in the design of ASME Code Class 1, 2, and 3 component supports in the CESSAR scope of design. Subsection NF of Section III of the ASME Code has been used in the design of these supports. The CESSAR design scope for specified component supports extends from the CE-supplied components to, but not including, the structural embedments. The pipe whip restraint design is within the BOP design scope and is discussed in Section 3.6.2 of this SER. However, any lugs welded to the reactor coolant piping system, whose primary function is to limit pipe motion due to a postulated pipe break, is within the CESSAR scope of design. Any anchor bolt preload requirements for CESSAR-supplied component supports are within the CESSAR design scope and are provided to the applicant of each plant referencing CESSAR in the detailed interface documents. We have audited the interface documents and find that the interface requirements are adequately defined to provide an acceptable design for the component supports.

We have reviewed the design procedure used for components supports in the CESSAR scope of design to Regulatory Guide 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports," and 1.130, "Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports." The two major subjects addressed by these guides are buckling of component supports and the design of bolts used in component supports.

We have reviewed the design procedures for buckling of component supports in the CESSAR scope of design. With respect to buckling of component supports we find the design procedures contained in CESSAR conform to the guidelines described above and are, therefore, acceptable. With respect to the design of bolts for component supports, stress limits for the design of bolts for all service conditions are provided in CESSAR. In particular, for faulted loading conditions, bolt tensile stresses are limited to the lesser of 0.7 Su or Sy at temperature, where Su and Sy correspond to the material ultimate and yield tensile stresses at temperature. Thus, we find the procedures used for the design of component supports in the CESSAR design scope to be acceptable.

A commitment is provided in CESSAR that the design criteria used for restraints provided solely to control the movement of postulated broken piping, in the CESSAR design scope, will be provided in the Safety Analysis Report that references CESSAR.

Based on our review of Section 3.9.3 of CESSAR our findings are as follows.

The specified design and service loading combinations used for the design of ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurances that, in the event of an earthquake or other service loading due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress or strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of component supports to withstand the most adverse combination of loading events without loss of structural integrity or supported component operability. The design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports constitute an acceptable basis for satisfying the applicable portions of GDC 1, 2, and 4 and, therefore, are acceptable.

3.9.3.1 Pump And Valve Operability Assurance

The staff has reviewed Section 3.9.3.2 of the CESSAR FSAR and compared the information provided with the Standard Review Plan guidelines. Based on our review we find that additional information is required in Section 3.9.3.2 of the CESSAR FSAR. We have discussed our findings with CE and they are in agreement that Section 3.9-3.2 should be amended to include the additional information. The purpose of this additional information is to provide a more definitive pump and valve operability assurance program, clarify the qualification methodology, and provide the acceptance criteria used in the program.

CE has indicated that the additional information can be submitted by November 16, 1981. We will review the information submitted and report the results of our findings in a revision to this report.

3.9.4 Control Rod Drive Systems

Our review, under Standard Review Plan Section 3.9.4, covers the mechanical design of the control element drive mechanism. We reviewed the analyses and tests performed to assure the structural integrity and functionality of the system during normal operations and under postulated accident conditions. We have also reviewed the life-cycle testing performed to demonstrate the reliability of the control element drive mechanism over its design life of 40 years.

Based on our review, we conclude the design criteria used and the testing program conducted in verification of the functionality of the control element drive mechanism are in conformance with Standard Review Plan 3.9.4. The use of these criteria provide reasonable assurance that the system will function reliably when required, and form an acceptable basis for satisfying the mechanical reliability requirements of GDC 27 and 29 and are, therefore, acceptable.

3.9.5 Reactor Pressure Vessel Internals

Our review under Standard Review Plan 3.9.5 is concerned with the load combinations, allowable stress limits and other criteria used in the design of the CE System 80 reactor internals. Reactor internals have been designed in accordance with Subsection NG, "Core Support Structures," of the ASME Code, Section III using the loads, load combinations, and allowable stress limits as provided in Section 3.9.3 of CESSAR. The description of the configuration and general arrangement of the reactor internal structures has been reviewed and found to be complete.

The specified transients, design and service loadings, and combination of loads as applied to the design of the CESSAR reactor internals provide reasonable assurance that in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these reactor internals would not exceed the allowable stress and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these reactor internals to withstand the most adverse loading events which have been postulated to occur during the service lifetime without loss of structural integrity or impairment of function. The design procedures and criteria used in the design of the CE System 80 reactor internals comply with Standard Review Plan 3.9.5 and constitute an acceptable basis for satisfying the applicable portions of GDC 1, 2, 4 and 10.

3.9.6 Inservice Testing of Pumps and Valves

In Sections 3.9.2 and 3.9.3 of this report, we discussed the design of safety-related pumps and valves. The load combinations and stress limits used in the design of pumps and valves provide assurance that the pressure integrity is maintained. However, to provide added assurance of the reliability of these components, the reference plants will be required to periodically test and perform periodic measurements of all its safety-related pumps and valves. These tests and measurements are to be performed in accordance with the rules of Section XI of the ASME Code and measurements are expected to verify that the pumps and valves will operate successfully when called upon. The periodic measurements are made of various parameters and compared to baseline measurements in order to detect long-term degradation of the pump or valve performance.

A requirement has been provided in CESSAR that this information will be provided in any Safety Analysis Report that references CESSAR. Specific review of this section will, therefore, be done on each reference plant application.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

The Seismic Qualification Review Team (SQRT), which consists of reviewers from the Equipment Qualification Branch and consultants from the Idaho National Engineering Laboratory (INEL), has reviewed the methodology and procedure of equipment seismic and dynamic qualification program contained in the pertinent CESSAR FSAR changes proposed by the CE as delineated in a letter from A. E. Scherer to G. Grimes, dated July 31, 1981 (LD-81-040). The CE Topical Report CENPD-182, entitled "Seismic Qualification of CE Instrumentation and Electric Equipment," was also reviewed. We find that these documents have, in general, defined the seismic and dynamic qualification program for NSSS seismic Category I mechanical and electrical equipment. CE is committed to the following position on the criteria of seismic and dynamic qualification of electrical equipment and instrumentation: For plants for which the CP application was docketed before October 27, 1972, the requirements of IEEE Std. 344-1971 will be met; for plants for which the CP application was docketed after October 27, 1972, the requirements and recommendations of IEEE Std. 344-1975 and the Regulatory Position of Regulatory Guide 1.100, which endorses IEEE Std. 344-1975, will be met. The qualification methodology used for BOP equipment will be evaluated in each reference plant application. The SQRT has concluded that the program described in the documents mentioned above meets the intent of the licensing criteria as described in IEEE 344-1975, Regulatory Guides 1.92 and 1.100, and the Standard Review Plan Sections 3.9.2 and 3.10.

A plant-specific onsite audit will be conducted and include a plant inspection to observe the as-built configuration and installation of the equipment. Furthermore, during the audit, the staff will review the qualification documentation (e.g., test report, analysis report, or a combination of both) of the equipment chosen for audit.

We conclude that the seismic qualification program outlined in CESSAR is acceptable. We will review the implementation of the CESSAR program, as well as the seismic qualification program and its implementation for the BOP scope, in the applications that reference CESSAR.

Applicants will be requested to provide information on the completion status of the equipment documentation, and onsite installation of the equipment. Before the audit is conducted, at least a 90% completion should be attained for both the equipment documentation and the onsite installation of the equipment.

3.11 Environmental Qualification for Safety-Related Electrical Equipment

In December 1979, the staff issued guidance for the environmental qualification of safety-related electrical equipment (NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"). The Commission Memorandum and Order (CLI-80-21), dated May 23, 1980) directs the staff to complete its review of environmental qualification including the publication of Safety Evaluation Reports for all Operator Reactors. In addition, this order directs that by no later than June 30, 1982, all electrical equipment in operating reactors subject to this review be in compliance with NUREG-0588 or the Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors.

By letter dated October 8, 1981, Combustion Engineering submitted a revised CESSAR FSAR text to make CESSAR consistent with the CE Topical Report, CENPD-255, Revision 2. The staff is currently reviewing the acceptability of the CESSAR environmental qualification (EQ) methods and procedures used to meet the requirements of IEEE Std 323-1974 and the "Category I" requirements of NUREG-0588; however, it is the responsibility of the reference plant applicant to review and evaluate the results of the EQ testing. The staff review of CESSAR and CENPD-255, Revision 2, for both "Mild" and "Harsh" environments is scheduled for completion by December 21, 1981, provided that CE is responsive to the staff's requests for additional information and that no significant issues arise during the course of the review.

4 REACTOR

4.1 Introduction

Criterion 10 of the General Design Criteria requires that the reactor core and associated systems be designed to asure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The staff reviewed the information provided in the Final Safety Analysis Report in support of the CESSAR reactor design. The staff evaluation is described below.

The CESSAR nuclear steam supply system is designed to operate at a maximum core thermal output of 3817 megawatts, with sufficient margin to allow for transient operation and instrument error, without causing damage to the core and without exceeding the pressure settings of the safety valves in the coolant system.

The reactor will be cooled and moderated by light water at a pressure of 2250 pounds per square inch, absolute. The reactor coolant will contain soluble boron for neutron absorption. The concentration of the boron will be varied, as required, to control relatively slow reactivity changes, including the effects of fuel burnup. Additional boron, in the form of burnable poison rods, will be employed to establish the desired initial reactivity. Part-length control element assemblies may be used for axial power shaping, and full-length control element assemblies will be used for reactor shutdown.

4.2 Fuel System Design

The review of the CESSAR fuel design which follows, is prepared in the format of the Standard Review Plan, Section 4.2. The objectives of this fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged" is defined as meaning that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements GDC 10 and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100 for postulated accidents. "Coolability," which is sometimes termed "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channel spacing to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the General Design Criteria (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accidents are given in 10 CFR Part 50.46.

To meet the above-stated objectives of the fuel system review, the following specific areas are critically examined: (a) design bases (and limits), (b) description and design drawings, (c) design evaluation, and (d) testing, inspection, and surveillance plans. In assessing the adequacy of the design, several items involving operating experience, prototype testing, and analytical predictions are weighed in terms of specific acceptance criteria for fuel system damage, fuel rod failure, and fuel coolability.

4.2.1 Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and suggest limiting values for important parameters such that damage will be limited to acceptable levels. For convenience, we group acceptance criteria for these design limits into three categories in the Standard Review Plan: (a) fuel system damage criteria, which apply to normal operation, including anticipated operational occurrences, (b) fuel rod failure criteria, which apply to normal operation, anticipated operational occurrences, and postulated accidents, and (c) fuel coolability criteria, which apply to postulated accidents.

4.2.1.1 Fuel System Damage Criteria

In the following paragraphs we review the design bases and corresponding design limits for the damage mechanisms listed in the Standard Review Plan.

(a) Design Stress

The design basis for fuel assembly, fuel rod, burnable poison rod, upper end fitting spring, and control element assembly (CEA) stresses is that the fuel system will be functional and will not be damaged due to excessive stresses.

The design limits for fuel assembly skeletal components for normal operation are as follows: The calculated general primary membrane stress (P_m) will be less than or equal to the design stress intensity (S_m) as defined by Section III of the ASME Boiler and Pressure Vessel code. Furthermore, the sum of P_m and the calculated primary bending stress (P_b) will be less than or equal to the product of S_m times the shape factor (F_s) of the cross section being analyzed. For cyclic loading conditions, the sum of the ratios of the number of cycles at a given stress condition to the maximum number permitted for that condition (i.e., the cumulative damage factor) will not exceed 0.8. The limit of 0.8 rather than 1.0 provides additional design margin.

The design limit for fuel rod and burnable poison rod cladding stress is that the maximum primary tensile stress is less than two thirds of the Zircaloy yield strength as affected by temperature.

The design limit for the Inconel X-750 upper end fitting spring is that the calculated shear stress will be less than or equal to the minimum yield stress in shear.

The design limits for the Inconel-625 clad CEAs are as follows: The calculated P_m will be less than or equal to S_m. For the Inconel cladding, the value of S_m is taken to be two thirds of the minimum yield strength as affected by temperature. Furthermore, the sum of P_m and P_b will be less than or equal to the product of F_s times S_m.

Many of these bases and limits are used by the industry at large. CE has employed various conservatisms in the limits such as the use of unirradiated yield strengths for zirconium-based alloys. We, therefore, conclude that the fuel assembly, fuel rod, burnable poison rod, upper end fitting spring, and CEA stress design bases and limits are acceptable.

(b) Design Strain

With regard to fuel assembly design strain, the design basis for normal operation is that permanent fuel assembly deflections shall not result in CEA insertion time beyond that allowable. This basis is satisfied by adherence to the stress criteria mentioned above.

For fuel rod and burnable poison rod cladding strain, a design limit for cladding circumferential plastic strain (due to cladding creep and pellet swelling) of 1% is employed as a means of precluding excessive cladding deformation. While we have not reviewed this design limit for normal operation, that value appears to be consistent with past practice and no numerical value for cladding strain during normal operation is provided as an acceptable criterion in the Standard Review Plan.

The design basis for CEA cladding strain is that the resultant dimensional clearances should be sufficient to allow CEA insertion within the required time. A strain limit of 1% ensures that the basis is satisfied. We find this design basis and limit to be acceptable.

(c) Strain Fatigue

The strain fatigue criterion is different from those described in SRP Section 4.2, viz., a safety factor of 2 on stress amplitude or of 20 on the number of cycles. Instead, cumulative strain cycling usage (i.e., the sum of the ratios of the number of cycles in a given effective strain range to the permitted number in that range) will not exceed 0.8. For Zircaloy cladding, the design limit curve has been adjusted to provide a strain margin for the effects of uncertainty and irradiation. The resulting curve bounds all of the data used in the development ("Fatigue Design Basis for Zircaloy Components," Nuclear Science and Engineering Volume 20, 1964) of the criterion that is discussed in the SRP. Therefore, the proposed criterion is acceptable.

(d) Fretting Wear

While the Standard Review Plan does not provide numerical bounding-value acceptance criteria for fretting wear, it does stipulate that the allowable fretting wear should be stated in the safety analysis report and that the stress and fatigue limits should presume the existence of this wear.

Significant fuel assembly fretting wear can occur in the uppermost portion of fuel assembly guide tubes where parked CEAs reside. The CESSAR fuel assembly design is expected to preclude such wear (see Section 4.2.3.1(d) of this report), hence there is no need for fuel assembly fretting wear design basis and limit.

CE does not use an explicit fretting wear limit in their stress and fatigue analysis for fuel and burnable poison rods. In view of relatively good operating experience with few resulting fuel and burnable poison rod failures, conservative stress analyses, and large fatigue margins, this design margin is acceptable.

The design basis and limit for CEA fretting wear are not given in the CESSAR FSAR. We will report our evaluation of this basis and limit in a revison to this report.

(e) Oxidation and Crud Buildup

Section 4.2 of the Standard Review Plan identifies cladding oxidation and crud buildup as potential fuel system damage mechanisms. General mechanical properties of the cladding are not significantly impacted by thin oxides or crud buildup. However, because of the increased thermal resistance of these layers, there is an increased potential for elevated temperature within the fuel as well as the cladding. Because the effect of oxidation and crud layers on fuel and cladding temperature is a function of several different parameters (e.g., heat flux and thermal-hydraulic boundary conditions), a design limit on oxide or crud layer thickness does not, per se, preclude fuel damage as a result of these layers and is not necessary. Rather, it is necessary that these layers be appropriately considered in other temperature-related fuel system damage and failure analyses (see Section 4.2.3.1(e) of this report).

(f) Rod Bowing

Fuel and burnable poison rod bowing are phenomena that alter the design-pitch dimensions between adjacent rods. Bowing affects local nuclear power peaking and the local heat transfer to the coolant. Rather than placing design limits on the amount of bowing that is permitted, the effects of bowing are included in the safety analysis. This is consistent with the Standard Review Plan and is acceptable. The methods used for predicting the degree of rod bowing are evaluated in Section 4.2.3.1(f), and the neutronic and thermal-hydraulic impact of the resulting bow are evaluated in Sections 4.3 and 4.4.

(g) Axial Growth

In the CESSAR design the core components requiring axial-dimensional evaluation are the CEAs, burnable poison rods, fuel rods, and fuel assemblies. The axial growth of the first of these components is primarily dependent upon the behavior of poison and spacer pellets and their interaction with the Inconel-625 cladding. The growth of the second two is mainly governed by (a) the behavior of poison, fuel, and spacer pellets, and their interaction with the Zircaloy-4 cladding and (b) the irradiation-and-stress (due to rod pressures being less than coolant pressure) -induced growth of the Zircaloy-4 cladding. The growth of the last is mostly a function of compressive creep of the Zircaloy-4 guide tubes and the irradiation-induced growth of the Zircaloy-4 guide tubes. For the Zircaloy cladding and fuel assembly guide tubes, the critical tolerances that require controlling are (a) the spacing between the fuel rods and the fuel assembly (i.e., shoulder gap) and (b) the spacing between the fuel assemblies and the core internals. Failure to adequately design for the former may result in fuel rod bowing, and for the latter may result in collapse of the holddown springs. With regard to inadequately designed shoulder gaps, problems have been reported in foreign (Obrigheim and Beznau) and domestic (Ginna) plants that have necessitated predischarge modifications to fuel assemblies.

The design basis and limit for CEA axial growth are not given in the FSAR. We will report our evaluation of this basis and limit in a revision to this report.

For fuel and burnable poison rods, allowances are made to ensure adequate (nonzero) shoulder gap clearance (at a 95% confidence level) to the upper fuel assembly end fitting such that the clearance is maintained throughout the design lifetime of the fuel. For fuel assembly axial growth, CE has a design basis that sufficient clearance between the fuel assembly and the upper guide structure exist throughout the expected lifetime of the fuel assembly. CE allocates a fuel assembly gap spacing, which will accommodate the maximum axial growth, when establishing the design minimum initial fuel assembly clearance with respect to the core internals. These above design bases and limits dealing with axial growth prevent mechanical interference and are thus acceptable.

(h) Rod Pressure

It is a mechanical design basis for core component rods that dimensional stability and cladding integrity are maintained. A necessary corollary of this design basis is that the driving force, rod internal pressure, is never so great as to result in loss of dimensional stability and cladding integrity.

Section 4.2 of the Standard Review plan identifies rod internal pressure as a potential fuel system damage mechanism. In this sense, damage is defined as an increased potential for elevated temperatures within the rod or an increased potential for cladding failure. Although the Standard Review Plan mentions only fuel and burnable poison rods, the mechanism also applies to CEAs. Because rod internal pressure is a driving force for, rather than a direct mechanism of, fuel system damage, it is not essential that a damage limit be specified. However, the Standard Review Plan presents an acceptance criterion that is sufficient in this regard and is widely used by the industry; it states that rod internal gas pressure should remain below the nominal system pressure during normal operation unless otherwise justified.

The CESSAR design criterion does not preclude fuel and burnable poison rod pressures from exceeding coolant system pressure and, thus, disagrees with the criterion in the Standard Review Plan. CE has not justified operation with fuel rod pressures exceeding system pressure. Therefore, the CESSAR fuel rod pressure design criterion is not acceptable, and we will report on the resolution of this issue in a revision to this report

For CEA rods, the rod internal pressure is limited such that the design strain limits are not exceeded. This, then, ensures scrammability within the required time. These limits are unchanged from previously approved CE fuel designs and remain acceptable for CESSAR.

(i) Assembly Liftoff

The Standard Review Plan calls for the fuel assembly holddown capability (gravity and springs) to exceed worst-case hydraulic loads for normal operation, which includes anticipated operational occurrences. The CESSAR FSAR endorses this design basis. We, therefore, conclude that the fuel assembly liftoff design basis is acceptable.

(j) Control Material Leaching

The Standard Review Plan and General Design Criteria require that reactivity control be maintained. Rod reactivity can sometimes be lost by leaching of certain poison materials if the cladding of control-bearing material has been breached.

The mechanical design basis for the CEAs is stated to be consistent with the loading considerations of Section III of the ASME Boiler and Pressure Vessel Code. Thus, the design basis for the CEA rods is to maintain cladding integrity; since cladding integrity would insure that reactivity is maintained, this design basis might appear to be acceptable. However, under some circumstances, unexpected breaches might go undetected, so we do not normally accept control rod cladding integrity as a sufficient design basis. A surveillance program that will confirm control rod reactivity will be required for plants that reference CESSAR.

4.2.1.2 Fuel Rod Failure Criteria

In the following paragraphs we review fuel rod failure thresholds for the failure mechanisms listed in the Standard Review Plan. When these failure thresholds are applied to normal operation including anticipated operational occurrences, they are used as limits (and hence SAFDLs) since fuel failure under those conditions should not occur according to the traditional conservative interpretation of GDC 10. When these thresholds are used for postulated accidents, fuel failures are permitted, but they must be accounted for in the dose calculations required by 10 CFR 100. The basis or reason for establishing these failure thresholds is thus established by GDC 10 and Part 100 and only the threshold values are reviewed below.

(a) Hydriding

Internal hydriding as a cladding failure mechanism is precluded by controlling the level of hydrogen impurities during fabrication. The moisture level in the uranium dioxide fuel is limited (to a proprietary value) by CE to less than 20 ppm, and this specification is compatible with the ASTM specification C776-76, Part 45 which allows 2 μgm hydrogen per gram of uranium (i.e., 2 ppm). This is the same as the limit described in the Standard Review Plan and, therefore, the more restrictive CE limit is acceptable.

A specific design basis and limit for external hydriding has been found unnecessary. As justification, CE has cited data that indicate (a) hydrogen absorption in Zircaloy cladding to be up to 250 ppm following 3 years of exposure and (b) acceptable burst ductility (i.e., 12%) of Zircaloy cladding containing 340 ppm hydrogen. Although we have not reviewed the CE references, we believe that such justification is reasonable and is consistent with that of other fuel vendors based on our experience. We are not aware of LWR fuel failures due to external hydriding. Therefore, we agree with CE that no design limit is required for external hydriding.

(b) Cladding Collapse

If axial gaps in the fuel pellet column were to occur due to densification, the cladding would have the potential of collapsing into a gap (i.e., flattening). Because of the large local strains that would result from collapse, the cladding is assumed to fail. It is a CE design basis that cladding collapse is precluded during the fuel rod and burnable poison rod design lifetime. This design basis is the same as that in the Standard Review Plan and is, therefore, acceptable.

(c) Overheating of Cladding

The design limit for the prevention of fuel failures due to overheating is that there will be at least 95% probability at a 95% confidence level that departure from nucleate boiling (DNB) will not occur on a fuel rod having the minimum DNBR during normal operation and anticipated operational occurrences. This design limit is consistent with the thermal margin criterion of SRP Section 4.2 and is, thus, acceptable. The specific DNBR limits and methods of analysis are evaluated in Section 4.4.2.

(d) Overheating of Fuel Pellets

As a second method of avoiding cladding failure due to overheating, CE avoids centerline fuel pellet melting as a design limit. This design limit is the same as given in the Standard Review Plan and is, thus, acceptable.

(e) Pellet/Cladding Interaction

As indicated in SRP Section 4.2, there are no generally applicable criteria for PCI failure. However, two acceptance criteria of limited application are presented in the SRP for PCI: (1) less than 1% transient-induced cladding strain and (2) no centerline fuel melting. Both of these limits are used in the CESSAR fuel design. Thus, the CESSAR design basis and limits agree with the only existing licensing criteria for PCI and is, therefore acceptable.

(f) Cladding Rupture

In the LOCA analysis of CESSAR plants, an empirical model is used to predict the occurrence of cladding rupture. The failure temperature is expressed as a function of differential pressure across the cladding wall. There are no specific design limits associated with cladding rupture, and the rupture model is an integral portion of the ECCS evaluation model, which is documented in the CE Topical Report CENPD-136.

(g) Mechanical Fracturing

The FSAR does not provide a discussion on the likelihood of fuel rod mechanical fracture that might be created by an externally applied force such as a hydraulic load or a load derived from core-plate motion. We will report on our evaluation of this issue in a revision to this report.

4.2.1.3 Fuel Coolability Criteria

For accidents in which severe fuel damage might occur, core coolability must be maintained as required by several GDC (e.g., GDC 27 and 35). In the following paragraphs we review limits that will assure that coolability is maintained for the severe damage mechanisms listed in the Standard Review Plan.

(a) Fragmentation of Embrittled Cladding

For LOCA analysis, CE uses the acceptance criteria of 2200°F on peak cladding temperature and 17% on maximum cladding oxidation as prescribed by 10 CFR 50.46. The FSAR does not provide a discussion on coolability criteria for other (i.e., non-LOCA) analysis. We will report on our evaluation of this issue in a revision to this report.

(b) Violent Expulsion of Fuel Material

In a CEA ejection accident, large and rapid deposition of energy in the fuel could result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal might be sufficient to destroy fuel cladding and the rod-bundle geometry and to provide significant pressure pulses in the primary system. To limit the effects of CEA ejection, the radially-averaged energy deposition at the hottest axial location is restricted to less than 280 cal/g. This limit is identical to the 280 cal/g limit given in Regulatory Guide 1.77 and is, therefore, acceptable.

(c) Cladding Ballooning and Flow Blockage

In the LOCA analyses of CESSAR plants, empirical models are used to predict the degree of cladding circumferential strain and assembly flow blockage at the time of hot-rod and hot-assembly burst. These models are each expressed as functions of differential pressure across the cladding wall. There are no specific design limits associated with ballooning and blockage, and the ballooning and blockage models are integral portions of the ECCS evaluation model, which is documented in the CE Topical Reports CENPD-136 and CENPD-133.

(d) Structural Damage from External Forces

To withstand the mechanical loads of a LOCA or an earthquake, the fuel assembly is designed to satisfy the stress criteria listed in Section 4.2.1.1(a), and guide-tube deformation is limited so as to not prevent CEA insertion during the safe shutdown earthquake. The maximum stress intensities for each individual event will be combined by a square root of the sum of the squares method. These criteria are similar to those described in Appendix A to Section 4.2 of the Standard Review Plan and are, therefore, acceptable.

4.2.2 Description and Design Drawings

The description of major fuel system components, including fuel rods, upper and lower end fittings, guide tubes, spacer grids, CEAs, and burnable poison rods is contained in the CESSAR FSAR. Numerical values and drawings are provided for various core components.

4.2.2.1 Design

The CESSAR System 80 core will be composed of 241 fuel assemblies and will employ a total of 89 CEAs, of which 13 CEAs will contain only a part-length poison column. Each CEA will consist of either 4 or 12 neutron absorber elements arranged to engage the peripheral guide tubes of fuel assemblies. Many design aspects of the CESSAR System 80 16x16 fuel assembly design are identical to those of previous CE NSSS plants such as Arkansas Nuclear One Unit 2 (ANO 2), San Onofre 2&3, Waterford Unit 3, and St. Lucie Unit 2. The most notable visual difference is in the design of upper and lower end fittings. The CESSAR active fuel zone will be 150 inches tall, and each fuel assembly will use ten Zircaloy-4 fuel rod spacer grids and one Inconel-625 bottom spacer grid.

While each parameter listed in SRP subsection 4.2.2 is not provided in the CESSAR FSAR, enough information is provided in sufficient detail for our review of the CESSAR design, and this information is acceptable.

4.2.2.2 Material Properties

The CESSAR FSAR provides or references various important material properties that are used in the CESSAR core analysis. We have reviewed these properties, which include such parameters as Young's Modulus of elasticity for Zircaloy cladding, thermal expansion of UO_2 pellets, melting point of Al_2O_3 pellets, thermal conductivity of Inconel-625 cladding, and helium release from B_4C pellets.

We have found that (a) the proposed properties are generally accepted by the nuclear industry at large, (b) the analysis is not particularly sensitive to the proposed properties, (c) the proposed properties have been explicitly approved in other documents such as FATES (CENPD-139-A) and remain acceptable, (d) the proposed properties are similar or conservative to properties that were proposed by other vendors and subsequently approved, (e) the proposed properties are a reasonable or conservative interpretation of publicly available data and correlations such as in MATPRO (NUREG/CR-0497, Rev.1), or (f) we have challenged the particular properties elsewhere and are pursuing resolution to the issue on a separate basis (see subsequent Sections 4.2.3.2(f) and 4.2.3.3(c) dealing with cladding swelling and rupture properties). We, therefore, conclude that material properties have been adequately addressed.

4.2.3 Design Evaluation

Section 4.2.1 presented design bases and design limits. In this section we review the CE methods of demonstrating that the CESSAR design meets the design criteria that have been established. This section will, therefore, correspond to Section 4.2.1 of this Safety Evaluation Report point by point. The methods of demonstrating that the design criteria have been met include operating experience, prototype testing, and analytical predictions.

4.2.3.1 Fuel System Damage Evaluation

(a) Design Stress

The FSAR does not provide the results of the stress analyses for the fuel assembly, fuel rod, burnable poison rod, upper end fitting spring, and CEA. We will report on our evaluation of these analyses in a revision to this report.

(b) Design Strain

The FSAR does not provide the results of the strain analyses for the fuel assembly, fuel rod, burnable poison rod, and CEA. We will report on our evaluation of these analyses in a revision to this report.

(c) Strain Fatigue

The FSAR does not provide the results of the strain fatigue analyses for the fuel assembly and fuel rod. We will report on our evaluation of these analyses in a revision to this report.

(d) Fretting Wear

Mechanical tests to demonstrate the effects of flow-induced vibration and consequent fatigue, fretting, and corrosion have been performed on 4x4 test assemblies and on full-sized 14x14, 15x15, and 16x16 fuel assemblies. In general, these tests adequately demonstrate that the effects of flow-induced vibration on the fuel rod are acceptable. However, a wear tendency that was not originally observed in the above-described flow tests has been found e.g., letters A. E. Scherer to V. Stello, dated December 23, 1977; W. P. Johnson to V. Stello, dated February 14, 1978; A. E. Lundvall to V. Stello, dated February 17, 1978 in irradiated fuel assemblies taken from operating CE reactors. These inspections detected unexpected degradation of guide tubes that are under CEAs. Coolant turbulence was responsible for inducing vibratory motions in the normally fully withdrawn CEAs and, when these vibrating control rods were in contact with the inner surface of the guide tubes, a wearing of the guide tube wall has taken place. The most substantial wear has been found to be limited to the relatively soft Zircaloy-4 guide tubes because the Inconel-625 cladding on the control rods provides a relatively hard wear surface. The extent of the observed wear has appeared to be plant dependent, but has in some cases extended completely through the guide tube walls.

The FSAR discussion on the propensity for guide tube wear in the CESSAR design was recently augmented by information contained in a letter from A. E. Scherer to J. R. Miller, dated October 2, 1981. As described in therein, the CESSAR core design will employ unique features that are different from those of previous CE NSSS plants and that are expected to alleviate guide tube wear.

The most significantly improved feature is better isolation of control rods where they enter the fuel assembly upper end fitting posts. In the region of the outlet plenum, each control rod is enclosed in a tubular structure, which also encases each associated post. This tubular structure is the lowermost portion of the upper guide structure and is composed of an array of tubes which join the CEA shrouds (located above the upper guide structure support plate) to the fuel assembly posts (located below the fuel alignment plate). Thus, each control rod element is shielded from crossflow effects (vortex shedding) of coolant exiting from the fuel assembly to the outlet plenum.

Another improved feature of the CESSAR design is in reduced guide tube flow rates. Flow testing in the CE hot loop flow facility, TF-2, has demonstrated that reduced bypass flow results in reduced control rod vibration. Consequently, CE has designed smaller fuel assembly guide tube flow holes for CESSAR System 80 than used in earlier CE NSSS plants.

In order to quantify the susceptibility of the CESSAR design to guide tube wear, fretting tests were conducted in TF-2 (see Appendix 4C, "Hot Loop Flow Testing of System 80 Fuel and CEA Components," of the CE response to NRC question 492.12). The TF-2 test arrangement consisted of an array of 5 dummy fuel assemblies, a 12-element CEA, and supporting structures which constitute a mockup of relevant reactor vessel internals. The tests were conducted at flow, temperature, and pressure conditions representative of the System 80 coolant operating conditions.

Following flow testing, eddy current examination of all rodded guide tubes was conducted. Those guide tubes which indicated the highest wear were then longitudinally sectioned to determine volumetric wear rates. The highest wear rate from this inspection was then used to project the maximum wear that could occur following three cycles of continuously rodded operation (a conservative assumption).

From fuel assembly stress analyses of lifting, holddown, seismic (operating basis earthquake), and seismic (safe shutdown earthquake) -plus-LOCA conditions, CE determined that (a) the latter was the limiting condition and (b) the maximum wear degradation predicted was about half as large as the allowable wear that would be acceptable for a CE plant with a large seismic ground acceleration.

It is CE's intent to verify actual inpile wear rates in the first System 80 plant (Palo Verde, Unit 1) during its first refueling outage. This surveillance program will investigate CEA integrity by measuring CEA cladding wear as well. At that time should guide tube wear rates be found unacceptable, the CESSAR guide tubes will be internally sleeved with chrome-plated, stainless-steel inserts similar to those approved on other CE NSSS plants such as Calvert Cliffs 1 & 2, Millstone 2, Arkansas Nuclear One Unit 2, St. Lucie 1 and 2, San Onofre 2 and 3, and Waterford 3.

Though the use of sleeve inserts has precluded guide tube wear, this remedy has adversely affected CEAs by exposing CEA cladding to a harder wear surface. Consequently, outage surveillance has been conducted or is planned at several CE facilities, and, as described in the CE Topical Report CENPD-225, in order to assess the degree of CEA fretting wear in those facilities and the ability to achieve the 10-year-design lifetime from CEAS. This need for CEA surveillance in CESSAR System 80 plants would need to be considered in the event sleeving is utilized.

With regard to the potential need for sleeving, the CESSAR design includes another new design feature that consists of an enlarged inner diameter of the guide tube and upper end fitting post. This enlarged region axially extends over the length wherein the sleeve insert would reside. This enlarged region would permit the insertion of a sleeve insert of inner diameter equivalent to the inner diameter of the non-enlarged guide tube region below the sleeve insert. Consequently, sleeving would not produce unacceptable scram times.

We view the new CESSAR design features as sensible alternatives to sleeving and they have been adequately verified by flow testing to permit whole-core applications. In the event that CE expectations are not confirmed by the surveillance program, CE has made a prudent modification that will permit remote installation of sleeve inserts which will not unacceptably protract control rod insertion times. We, therefore, conclude that guide tube fretting wear has been adequately addressed provided that the first CESSAR System 80 applicant commit to the above described surveillance program.

With regard to the CE fretting analysis of the fuel cladding, we conclude the following:

1. Out-of-pile flow test to determine the adverse effects of fretting wear that is anticipated for the 16x16 fuel design demonstrate the acceptability of the CESSAR design.

2. Light water reactor operating experience demonstrates that the number of fretting-induced fuel failures is insignificant.

3. There is only a small dependence of cladding stresses on fretting wear because this type of wear is local at grid-contact locations and relatively shallow in depth.

4. The built-in conservatisms in the stress and strain fatigue limits adequately offset the effect of fretting wear degradation.

Therefore, we conclude that the CESSAR fuel rods will perform adequately with respect to fretting wear.

(e) Oxidation and Crud Buildup

The CESSAR FSAR states that, during normal and upset conditions, oxidation and crud buildup have not been observed as a problem. Presumably this statement holds true provided that the CE recommended coolant chemistry specifications (pH, oxygen, lithium, etc.) are maintained. We find that this issue has been adequately addressed for nonextended burnup cycles, and we are pursuing this issue generically with CE for high burnup schemes.

(f) Rod Bowing

The consideration of both fuel and poison rod bowing in the 16x16 design was documented in the topical report CENPD-225, "Fuel and Poison Rod Bowing," and CESSAR references this report.

However, CENPD-225 will not be approved prior to June 1982. For interim acceptance of methods by which rod bowing analyses can be made, the staff has issued three reports* in which we have (a) given approval of a burnup-dependent approach to rod bowing, (b) presented a formulation to be used in extrapolating bow magnitudes to new designs (i.e., 16x16), and (c) described a factor that increases the cold rod bow magnitudes (which are determined from cold-measured gap closures in spent fuel pools) to account for hot rod bow magnitudes that occur in-reactor during hot-operating conditions. These approved methods should be used in determining fuel rod bowing penalties for CESSAR.

To that end, we are requiring (see response to NRC question 492.3) that CE (a) amend the CESSAR FSAR to incorporate the interim correlation for the burnup-dependent prediction of rod bowing magnitude, (b) identify in the base to the Technical Specifications plant-specific or generic margins (credits) used to offset the reduction in DNBR due to fuel rod bowing, and (c) incorporate the residual rod bowing penalty (see Section 4.4.3.1 of this SER) into the Technical Specifications. (It is expected that the CESSAR fuel will experience rod bowing equal to that predicted for St. Lucie 2. This is because of the similarity of spacer grid span lengths and fuel rod cladding dimensions.)

(g) Axial Growth

The FSAR does not provide the results of the axial growth analysis for the CEA. We will report on our evaluation of this analysis in a revision to this report.

The CESSAR FSAR references a CE topical report CENPD-198, "Zircaloy Growth In-Reactor Dimensional Changes in Zircaloy-4 Fuel Assemblies," in support of a discussion on the dimensional stability of Zircaloy components. The report accounts for differences in growth, fabrication tolerances, elastic compression, creep, and thermal expansion between the cladding and the guide tubes. We have reviewed this topical report and supplements but our approval, in a letter to CE dated August 21, 1979, was limited to an axially averaged fast neutron fluence of $4x10^{21}$ n/cm^2, which corresponds to a maximum assembly exposure of 22.5 GWd/t. This is an exposure above which CE has not reported data on their core components but below the design exposures planned for CESSAR plants.

Assurance on the acceptability of the CESSAR fuel design beyond an exposure of 22.5 GWd/t will be available from reference plant applicants' routine visual fuel assembly inspection programs, which will be performed during or following each refueling outage. Thus any trend toward unanticipated growth or mechanical interference will be evident during inspection. In addition, during the first three refueling outages of Arkansas Nuclear One Unit 2 (a plant whose fuel design was also based on the CENPD-198 methods), the length of the fuel assembly and peripheral fuel rods will be precisely measured in six assemblies (two from each

*Memoranda: D. F. Ross and D. G. Eisenhut to D. B. Vassallo and K. R. Goller, "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors," dated December 8, 1981; D. F. Ross, and D. G. Eisenhut to D. B. Vassallo and K. R. Goller, "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors," dated February 16, 1977; and R. O. Meyer to D. F. Ross, "Revised Coefficients for Interim Rod Bowing Analysis," dated March 2, 1978.

fuel region) that have been extensively precharacterized as described in the FSAR. Thus, we will be able to compare the measured values with those calculated as the burnup progresses. If a noncorservative gap closure is observed, remedial action can be taken before safety is affected.

In addition, we find the CE method of explicitly accommodating predicted fuel assembly growth in the core internals design to be acceptable.

(h) Rod Pressure

The analysis of fuel rod internal pressure for the CESSAR fuel design was performed with the fuel performance code, FATES, which has been approved by the staff. However, we have questioned in NUREG-0418 the validity of fission gas release calculations in most fuel performance codes including FATES for burnups greater than 20 GWd/t. CE was informed of this concern, and NUREG-0418 provided a method of correcting gas release calculations for burnups greater than 20 GWd/t. Since there was no question of the adequacy of FATES for burnups below 20 GWd/t, the CESSAR calculations will be acceptable for operation early in life until the peak local burnup reaches 20 GWd/t. For burnups in excess of that value, FATES calculations (and other affected analyses) will have to be redone using the correction method mentioned above or such modified methods that might be submitted by CE or an applicant and approved by NRC.

We will accordingly condition the CESSAR applicants' operating licenses (in a like manner as done previously for Arkansas Nuclear One Unit 2, San Onofre Unit 2, Waterford Unit 3, and St. Lucie Unit 2) to require resolution of this issue prior to the cycle in which each CESSAR System 80 core achieves a peak pellet exposure of 20 GWd/t.

For burnable poison rods, the pressure analysis is incomplete. We will report on our evaluation of this analysis in a later report.

(i) Assembly Liftoff

The CESSAR System 80 fuel assembly liftoff analysis is in progress at CE. We will report on our evaluation of this analysis in a revision to this report.

(j) Control Material Leaching

While the design basis for the CEAs is to maintain cladding integrity, and while the probability of CEA cladding failure appears to be quite low, we have considered the corrosion behavior of the control material (B_4C and Inconel) in the postulated case of CEA cladding perforation or failure. We believe that the control materials in their physical states of application are relatively inert, and it would take several months for a significant boron loss to occur. However, the rods held in safety banks would not normally have their reactivity worth routinely assessed because they would not normally be used. Therefore, as a routine matter, we require licensees who use B_4C-filled rods to perform surveillance designed to assure that the reactivity invested in the rods is not being lost through a cladding defect.

In Section 14.2.12.4.2 of the CESSAR FSAR, a CEA rod symmetry test that can detect CEA failures is discussed. The symmetry test is sensitive enough to detect the loss of substantial reactivity from any single CEA. The CEA testing will be performed during initial low-power physics testing. We believe that the CEA symmetry test, if performed not only for initial startup but at the beginning of each cycle after fuel loading, will adequately confirm CEA integrity. Arrangements for this or alternative testing will be made with individual CESSAR applicants.

Based on the facts that (a) the CEA rod design should preclude failure, (b) even if cladding failure should occur, the corrosion rate of the B_4C poison material is low enough to require several months for significant loss to occur, and (c) reactivity checks will be performed after each refueling to verify the adequacy of the CEA rod design, we conclude that the issue of CEA poison material leaching has been adequately addressed, provided the reference plant applications include an acceptable CEA surveillance program.

4.2.3.2 Fuel Rod Failure Evaluation

(a) Hydriding

CE adheres to the moisture limits described in SRP subsection 4.2.1.2 and has not reported fuel failures due to hydriding in CE operating reactors. We conclude, therefore, that reasonable evidence has been provided that hydriding as a fuel failure mechanism will not be significant in a CESSAR core.

Some CE burnable poison rods have experienced failures due to primary hydriding, as described in the CE Topical Report CEN-77 (M)-P. Subsequently, CE made changes to the burnable poison rod design and manufacturing processes. The changes included reduced pellet moisture limit and revised manufacturing processes aimed at reducing moisture ingress to the poison rod. We approved these modifications on earlier CE NSSS plants and agree that they reduce the potential for primary hydriding of burnable poison rods. No further failures of this kind have been reported.

(b) Cladding Collapse

In calculating the time at which cladding collapse will occur, CE uses the generic methods described in CEPAN Topical Report CENPD-187-A which is approved for licensing applications. Inputs to the analysis include minimum fill gas pressurization, worst-case combination of manufacturing tolerances, no fission gas release, conservative flux, and temperature histories, and the analysis does not reply on the support of pellets or plenum springs. CE adjusts rod pressure such that cladding collapse will not be calculated to occur at a residence time that is less than that of the CESSAR design lifetime. This calculation has not been completed for the CESSAR System 80 plants; however, CE has agreed to submit such a generic analysis. We will report on this issue in a revision to this report.

(c) Overheating of Cladding

As stated in SRP Section 4.2, adequate cooling is assumed to exist when the thermal margin criterion to limit the departure from nucleate boiling (DNB) or boiling transition in the core is satisfied. The method employed to meet the DNB design basis is evaluated in Section 4.4.2 and will not be discussed here.

(d) Overheating of Fuel Pellets

The design evaluation of the fuel centerline melt limit is performed with the approved CE fuel performance code, FATES. This code is also used to calculate initial conditions for transients and accidents described in Chapter 15 of the Standard Review Plan and this report.

In applying the FATES code to the centerline melting analysis, the melting temperature of the UO_2 is assumed to be 5080°F unirradiated and is decreased by 58°F per 10 GWd/t. This relation has been almost universally adopted by the industry and has been accepted by the NRC staff in the past.

The FSAR states that the CESSAR fuel analysis does not predict incipient fuel centerline melting for normal operation and transients. Therefore, this design limit is satisfied.

(e) Pellet/Cladding Interaction

The only two PCI criteria in current use in licensing (1% cladding strain and no fuel melting), while not broadly applicable, are easily satisfied. CE uses FATES to calculate creep strain, and the values calculated by that code are found to be below the 1% strain criterion. And, as indicated in the discussion on overheating failures, the no-centerline-melt criterion is satisfied. Therefore, the two existing licensing criteria for PCI have been satisfied.

The burnable poison rods are designed to ensure hermeticity for all normal operating and anticipated transient conditions. In burnable poison rods, the reactivity control material is Al_2O_3/B_4C pellets, which is appreciably susceptible to leaching. Thus, burnable poison rod cladding integrity is essential.

The effects of pellet/cladding interaction have been observed in burnable poison rods as described in the CE Topical Report CEN-50. In burnable poison rods, pellet/cladding interaction has predominately resulted in excessive axial growth of the rod, rather than perforation of the cladding wall. To reduce the potential for burnable poison rod growth, CE has made several pertinent modifications and manufacturing process changes. These revisions consist of the following: (a) increased pellet-to-cladding gap, (b) chamferred pellets, (c) increased rod pressurization, and (d) reduced plenum spring preload. We have reviewed these revisions on CE NSSS plant operating license applications and agree that they should significantly reduce pellet/cladding interaction and the potential for burnable poison rod failure.

(f) Cladding Rupture

Although the cladding rupture temperature model is an integral part of the approved ECCS evaluation model, we have concluded that the model is nonconservative over some regions of applicability as described in NUREG-0630. Therefore, until this issue is generically resolved, supplemental calculations will be required to accompany the CE ECCS evaluation model results for CESSAR. These supplemental calculations should demonstrate that CESSAR would conform to the ECCS acceptance criteria of 10 CFR 50.46 if the NRC staff rupture temperature correlation in NUREG-0630 was substituted for the CE model contained in CENPD-136.

This requirement for supplemental ECCS calculations is the same as the present requirement made for all operating license applications and all ECCS reanalyses of operating reactors.* (See the following paragraph (c) of Section 4.2.3.3 for a concurrent requirement on cladding ballooning and flow blockage models.) For these supplemental calculations only, we accept other compensatory model changes that may not yet be approved by the NRC, but are consistent with the changes allowed for the confirmatory operating reactor calculations mentioned above. This analysis has not been provided; however, we have been advised by CE that these analysis are essentially complete, demonstrate acceptable results, and are currently being reviewed by CE prior to documentation in a revision to this report.

The overall impact of cladding rupture on the response of the CESSAR design to the loss-of-coolant accident is discussed in Section 15.3.8 and not reviewed further in this section.

(g) Mechanical Fracturing

The FSAR has not provided the results of the fuel rod mechanical fracture analysis. We will report on our evaluation of this issue in a report.

4.2.3.3 Fuel Coolability Evaluation

(a) Fragmentation of Embrittled Cladding

The primary degrading effect of a significant degree of cladding oxidation is embrittlement of the cladding. Such embrittled cladding will have a reduced ductility and resistance to fragmentation. The most severe occurrence of such embrittlement is during a LOCA. The overall effects of cladding embrittlement on the CESSAR design for the loss-of-coolant accident are discussed in Section 15.3.8 and are not reviewed further in this section.

One of the analytical methods that is used to provide input to the analysis described in Section 15.3.8 is the steady-state fuel performance code, FATES. This code provides fuel pellet temperatures (stored energy) and fuel rod gas inventories for the ECCS evaluation model as prescribed by Appendix K to 10 CFR Part 50. The code accounts for fuel thermal conductivity, fuel densification, gap conductance, fuel swelling, cladding creep, and other phenomena that affect the initial stored energy. Although the FATES code was approved by NRC, its validity at high burnups has been questioned. The FATES analysis is therefore accepted for LOCA analysis only for burnups less than 20 GWd/t, and reanalysis with approved methods will be required (see previous discussion in Section 4.2.3.1(h)) for applications involving higher burnups.

The FSAR does not provide coolability analyses of non-LOCA events that involve high cladding temperatures (DNB events). We will report on our evaluation of this issue in a later report.

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*Letter from D. G. Eisenhut to All Operating Light Water Reactors, dated November 9, 1979, and a memorandum from H. R. Denton to the Commissioners, dated November 26, 1979.

(b) Violent Expulsion of Fuel Material

The analysis that demonstrates that the design limit is met for this event for the CESSAR design is presented in Section 15.4.5 of the CESSAR FSAR and is reviewed in Section 15.2.4.7 of this report.

(c) Cladding Ballooning and Flow Blockage

Although the cladding ballooning and assembly flow blockage models were approved as integral parts of the ECCS evaluation model, we have concluded in NUREG-0630 that both models are nonconservative over some regions of applicability. Therefore, until this issue is generically resolved, supplemental calculations will be required for each plant application that uses the CE ECCS evaluation model. These supplemental calculations should demonstrate that each plant would conform to the ECCS acceptance criteria of 10 CFR 50.46 if the NRC staff cladding strain and assembly flow blockage models in NUREG-0630 were substituted for the CE models contained in CENPD-136 and CENPD-133.

This requirement for supplemental ECCS calculations is the same as the present requirement made for all operating license applications and all ECCS reanalyses of operating reactors. (See paragraph (f) of Section 4.2.3.2 for a concurrent requirement on the cladding rupture model and references). For these supplemental calculations only, we accept other compensatory model changes that may not yet be approved by the NRC, but are consistent with the changes allowed for the confirmatory operating reactor calculations mentioned above. This analysis has not yet been provided; however, we have been advised by CE that these analyses are essentially complete, demonstrate acceptable results, and are currently being reviewed by CE prior to documentation. We will report on the resolution of this issue in a revision to this report. The overall impact of cladding ballooning and assembly flow blockage models on the response of the CESSAR design to the loss-of-coolant accident is evaluated in Section 15.3.8 and is not reviewed further in this section.

(d) Structural Damage from External Forces

For this analysis, the CESSAR FSAR references the topical report CENPD-178, "Structural Analysis of Fuel Assemblies for Combined Seismic and Loss-of-Coolant Accident Loading." This topical report was not accepted for CESSAR calculations because (a) it is not a generic report, but rather only a plant-specific report for Arkansas Nuclear One Unit 2 (see response to NRC question 231.26) in and (b) the NRC staff review of CENPD-178 concluded that the report did not contain an adequate model for analyzing lateral loads on the fuel assembly and there was insufficient information on spacer grid tests, as described in a letter to CE dated February 2, 1978.

As a result of the earlier review of CENPD-178, CE has submitted Revision 1 to CENPD-178. We have not completed our review of this revised report, but that review is scheduled for completion by January 15, 1982. The revised version of the CESSAR FSAR has been amended to reference the revised version of CENPD-178. Because the input requires plant-specific data, this analysis must be submitted on individual plant applications which make reference to CESSAR as the reference fuel system design.

4.2.4 Testing, Inspection, and Surveillance Plans

4.2.4.1 Testing and Inspection of New Fuel

As described in SRP Section 4.2, testing and inspection plans for new fuel should include verification of significant fuel design parameters. While details of the manufacturer's testing and inspection programs should be documented in quality control reports, the programs for fabrication and onsite inspection of new fuel, core components, and CEAs should also be described in the FSAR.

The CESSAR FSAR discussion of the CE quality assurance program addresses tolerance requirements for fuel system components and parts, density specifications for fuel pellets, and rod inspection techniques, alignment verification on CEAs, acceptance standards for welds, etc. Fuel system component inspections rely on both nondestructive and destructive examinations. Fuel pellet inspections, for example, are performed to establish parameters such as enrichment, density, length, and surface roughness. Fuel rod inspections, consist of various nondestructive examination techniques such as leak testing, weld inspection, and fluoroscopy.

We conclude, based on the information provided in Section 4.2.4 of the FSAR, that the new-fuel testing and inspection program for the CESSAR fuel system is acceptable.

4.2.4.2 On-Line Fuel System Monitoring

Section 9.3.4.5.6.1 of the CESSAR FSAR describes the process radiation monitor. This is a scintillation detector that taps into the primary coolant system at the purification letdown line and provides a continuous signal to a ratemeter that is located in the control room. The analyzer utilizes gamma-ray spectrometry to monitor gross gamma and specific fission product (i.e., high yield isotopes such as rubidium-88) gamma activity in the reactor coolant. The ratemeter is equipped with a variable-setpoint, high-level alarm. The FSAR states that increasing trends in fission product activity will be interpreted as indications of fuel failures. Confirmation, however, will be obtained by chemical analysis of primary coolant samples via the procedures described in the "Chemical and Volume Control System," Section 9.3.4 of the CESSAR FSAR.

We conclude that the CESSAR on-line fuel failure monitoring system and the intended use of the system are acceptable.

4.2.4.3 Post-Irradiation Surveillance

CE has instituted a fuel surveillance program for the 16x16 fueled reactor core. This program is being conducted in Arkansas Nuclear One Unit 2 and involves the irradiation of six standard 16x16 fuel assemblies--two in each fuel loading region. Each assembly includes a minimum of 50 precharacterized, removable rods. Interim examination of all remaining test assemblies will be conducted during the first three refueling outages.

The staff concludes that the design-oriented surveillance program originally proposed by CE will adequately demonstrate the performance of the 16x16 fuel assembly, if that program is supplemented with a more comprehensive, but less detailed, surveillance program at each individual plant.

According to the Standard Review Plan, a minimum acceptable supplemental program should include a qualitative visual examination of some discharged fuel assemblies after each refueling. Such a program should be sufficient to identify gross problems of structural integrity, fuel rod failure, fuel rod bowing, spacer grid strap damage, insufficient fuel rod shoulder gap spacing, or crud deposition. There should also be a commitment in the program to notify NRC and perform additional surveillance if unusual behavior is noticed in the visual examination or if the plant process radiation monitor indicates gross fuel failures. We will review the surveillance program in each reference plant application.

4.2.5 Evaluation Findings

One open issue that must yet be resolved is the rod pressure limit (see paragraph 4.2.1.1(h)).

Various fuel design calculations and analyses of lesser importance (see 4.2.1.1(d and g), 4.2.1.2(g), 4.2.1.3(a and d), 4.2.3.1(a, b, c, g, h, and i), 4.2.3.2(g), and 4.2.3.3(a) are also incomplete. We have combined these calculations and analyses and consider them as one confirmatory issue. In addition, the cladding collapse analysis (4.2.3.2(b)) and the supplementary ECCS calculations (4.2.3.2(f) and 4.2.3.3(c)) are incomplete and are considered confirmatory issues.

The issue of high-burnup fission gas release enhancement is important, but need not be resolved for issuance of the FDA, provided that licenses for reference plants are conditioned to require resolution prior to the cycle in which each core attains the specified burnup limitation (see paragraph 4.2.3.1(h)).

When the above-described issues are resolved, we will conclude that the CESSAR fuel system has been designed so that (a) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (b) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (c) core coolability will always be maintained, even after severe postulated accidents, and thereby meets the related requirements of 10 CFR Part 50.46; 10 CFR Part 50, Appendix A, General Design Criteria 10, 27, and 35; 10 CFR Part 50, Appendix K; and 10 CFR Part 100. This conclusion is based on the following:

1. CE has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with control rod ejection and fuel densification have been performed in accordance with the guidance of Regulatory Guide 1.77 and with an acceptable alternative to Regulatory Guide 1.126.

2. CE has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. We will require that reference plant applicants make commitments to perform CEA reactivity checks, and post-irradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

3. CE has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated and thereby meet the related requirements of 10 CFR Part 100. In meeting these requirements CE has (a) used the fission-product release assumptions of Regulatory Guides 1.25 and 1.77, and an acceptable (more conservative) alternative to Regulatory Guide 1.4 and (b) performed the analysis for fuel rod failures for the rod ejection accident in accordance with the guidelines of Regulatory Guide 1.77.

On the basis of our review of the fuel system design, we conclude that the CESSAR fuel system design has met all the requirements of the applicable regulations, regulatory guides, and current regulatory positions with the exceptions noted above.

All applicants referencing the CESSAR FSAR will be required to supply following plant-specific information.

1. A CEA surveillance program (see paragraph 4.2.1.1(j)).

2. A fuel assembly guide tube fretting wear program (see paragraph 4.2.3.1(d)). This requirement pertains only to the first CESSAR applicant.

3. A fuel assembly loads analysis due to combined seismic - and - LOCA (see paragraph 4.2.3.3(d)).

4. A commitment to perform a general fuel surveillance program (see paragraph 4.2.4.3).

4.2.6 <u>References</u>

1. W. J. O'Donnel and B. F. Langer, "Fatigue Design Basis for Zircaloy Components," <u>Nuc</u>. <u>Sci</u>. <u>Eng</u>., Vol. 20, p. 1 (1964).

2. H. Schenk, "Experience from Fuel Performance at KWO," International Atomic Energy Report SM-178-15, October 1973.

3. K. Kuffer and H. R. Lutz, "Experience of Commercial Power Plant Operation in Switzerland," Fifth Foratom Conference, Florence, Italy (1973).

4. Rochester Gas and Electric Corporation, "Robert Emmett Ginna, Nuclear Power Plant, Unit 1, Final Safety Analysis Report," Docket Number 50-244, p. 103, 1972.

5. Standard Specifications for Sintered Uranium Dioxide Pellets," ASTM Standard C776-76, Part 45 (1977).

6. "High Temperature Properties of Zircaloy and UO_2 for Use in LOCA Evaluation Models," CE Report CENPD-136, July 1974.

7. "CEFLASH-4A, A Fortran IV Digital Computer Program for Reactor Blowdown Analysis," CE Report CENPD-133, April 1974.

8. "Fuel Evaluation Model," CE Report CENPD-139-A, July 1974.

9. "MATPRO-Version 11 (Revision 1): A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," NRC Report NUREG/CR-0497, Rev. 1, February 1980.

10. Letter from A. E. Scherer, CE, to V. Stello, NRC, LD-77-122, December 23, 1977.

11. Letter from W. P. Johnson, MYAPCo, to V. Stello, NRC, February 14, 1978.

12. Letter from A. E. Lundvall, BG&ECo, to V. Stello, NRC, February 17, 1978.

13. Letter from A. E. Scherer, CE, to J. R. Miller, NRC, LD-81-066, October 2, 1981.

14. Letter from W. G. Counsil, NU, to R. Reid, NRC, Subject: Sleeved CEA Guide Tube Inspection Program, March 29, 1979.

15. Letter from W. G. Counsil, NU, to R. Ried, NRC, Subject: CEA Guide Tube Inspection Program, April 17, 1979.

16. Letter from R. C. L. Olson, BG&ECo, to R. W. Reid, NRC, Subject: CEA Guide Tube Inspection Program, October 10, 1979.

17. Letter from R. H. Groce, MYAPCo, to R. W. Reid, NRC, Subject: Questions Pertaining to Cycle 5 Reload Fuel, November 6, 1979.

18. Letter from W. G. Counsil, NU, to R. Reid, NRC, Subject: Resolution of Cycle 3 Startup Commitments, April 15, 1980.

19. Letter from W. F. Lee and W. G. Counsil, NU, to R. A. Clark, NRC, Subject: Resolution of Cycle 3 Startup Commitments, August 14, 1980.

20. Letter from A. E. Lundvall, Jr., BG&ECo, to R. A. Clark, NRC, Subject: Responses to NRC Staff Questions, November 20, 1980.

21. Letter from A. E. Lundvall, Jr., BG&ECo, to R. A. Clark, NRC, Subject: CEA Guide Tube Inspection Program, January 21, 1981.

22. Letter from A. E. Lundvall, Jr., BG&ECo, to R. A. Clark, NRC, Subject: CEA Guide Tube Inspection Program, April 10, 1981.

23. Letter from D. C. Trimble, AP&LCo, to R. A. Clark, NRC, Subject: Preliminary Results of ANO-2 Fuel Inspection, May 22, 1981.

24. "Fuel and Poison Rod Bowing," CE Report CENPD-225, October 1976.

25. Memorandum from D. F. Ross and D. G. Eisenhut, NRC, to D. B. Vassallo and K. R. Goller, "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors," December 8, 1976.

26. Memorandum from D. F. Ross and D. G. Eisenhut, NRC, to D. B. Vasallo and K. R. Goller, "Revised Interim Safety Evaluation Report on the Effects of

Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors," February 16, 1977.

27. Memorandum from R. O. Meyer, NRC, to D. F. Ross, "Revised Coefficients for Interim Rod Bowing Analysis," March 2, 1978.

28. "Zircaloy Growth: In-Reactor Dimensional Changes in Zircaloy-4 Fuel Assemblies," CE Report CENPD-198, December 1975

29. "Zircaloy Growth: Application of Zircaloy Irradiation Growth Correlations for the Calculations of Fuel Assembly and Fuel Rod Growth Allowances," CE Report CENPD-198, Supplement 1, December 1977.

30. "Response to Request for Additional Information on CENPD-198-P, Supplement 1," CE Report CENPD-198, Supplement 2-P, November 1, 1978.

31. Letter from R. L. Baer, NRC, to A. E. Scherer, CE, August 21, 1979.

32. Arkansas Power and Light Company, "Arkansas Nuclear One, Unit 2, Final Safety Analysis Report," Docket Number 50-368, p. 4.2-7a, May 25, 1977.

33. N. Fuhrman and D. B. Scott, "Cladding Damage Analysis of Maine Yankee Core II Burnable Poison Rods," CE Report CEN-77 (M)-P, January 1978.

34. "CEPAN Method of Analyzing Creep Collapse of Oval Cladding," CE Report CENPD-187-A, March 1976.

35. "CE Burnable Poison Irradiation Test Program," CE Report CEN-50, March 1977.

36. Letter from D. G. Eisenhut, NRC, to all Operating Light Water Reactors, November 9, 1979.

37. Memorandum from H. Denton, NRC, to Commissioners, "Potential Deficiencies in ECCS Evaluation Models," November 26, 1979.

38. "Structural Analysis of Fuel Assemblies for Combined Seismic and Loss-of-Coolant Accident Loading," CE Report CENPD-178, August 1976.

39. "San Onofre Nuclear Generating Station, Units 2 and 3, Final Safety Analysis Report," SCECo and SDG&ECo, Amendment 16, p. Q&R 4.2-27, Docket Numbers 50-361/362, September 1979.

40. Letter from D. F. Ross, Jr., NRC, to A. E. Scherer, CE, February 2, 1978.

41. Letter from A. E. Scherer, CE, to J. R. Miller, NRC, Subject: CENPD-178-P, Rev. 1-P, "Structural Analysis of Fuel Assemblies for Seismic and Loss-of-Coolant Accident Loading," August 21, 1981.

4.3 Nuclear Design

The nuclear design of the System 80 reactor is in many respects similar to that of the CE 3390 thermal megawatt reactor design used by San Onofre Units 2 and 3 and Waterford Unit 3. The principal difference is an increase in the number of fuel assemblies (217 to 241) which results in a higher total reactor power of 3800 MWt.

Our review of the nuclear design was based on information contained in the FSAR, amendments thereto, and the referenced topical reports. Our review was conducted in accordance with the guidelines provided by the Standard Review Plan, Section 4.3.

4.3.1 Design Basis

Design bases are presented which comply with the applicable General Design Criteria. Acceptable fuel design limits are specified (GDC 10), a negative prompt feedback coefficient is specified (GDC 11) and tendency toward divergent operation (power oscillation) is not permitted (GDC 12). Design bases are presented which require a control and monitoring system (GDC 13) which automatically initiates a rapid reactivity insertion to prevent exceeding fuel design limits in normal operation or anticipated transients (GDC 20). The control system is required to be designed so that a single malfunction or single operator error will cause no violation of fuel design limits (GDC 25). A reactor coolant boration system is provided which is capable of bringing the reactor to cold shutdown conditions (GDC 26) and the control system is required to control reactivity changes during accident conditions when combined with the engineered safety features (GDC 27). Reactivity accident conditions are required to be limited so that no damage to the reactor coolant system boundary occurs (GDC 28).

We find the design bases presented in the FSAR to be acceptable.

4.3.2 Design Description

The FSAR contains the description of the first cycle fuel loading which consists of three different enrichments and has a first cycle core average burnup of 16576 MWD/T. Fuel enrichment and burnable poison distributions are shown. Assembly enrichments, core burnup, critical soluble boron concentrations and worths, and plutonium buildup are also presented. Values presented for the delayed neutron fraction and prompt neutron lifetime at beginning and end of cycle are consistent with those normally used and are acceptable.

Power Distribution

The design bases affecting power distribution are:

- The peaking factor in the core will not be greater than 2.28 during normal operation at full power in order to meet the initial conditions assumed in the loss-of-coolant accident analysis.

- Under normal conditions (including maximum overpower) the peak fuel power will not produce fuel centerline melting.

- The core will not operate during normal operation or anticipated operational occurrences, with a power distribution that will cause the departure from nucleate boiling ratio to fall below 1.19 (using the CE-1 DNBR correlation).

The applicants plan to employ a reactor monitoring system, designated the core operating limit supervisory system (COLSS), to continuously monitor important reactor characteristics and establish margins to operating limits. This system, which consists of software executed on the plant computer, will utilize the output of the incore detector system to synthesize the core average axial power distribution. Rod positions taken from the control rod position indication system, together with precalculated radial peaking factors, will be used to construct axially dependent, radial power distributions. By using this information, together with measured primary coolant flow, pressure, and temperature, the core operating limit supervisory system will establish the margin to the operating limits on maximum linear heat generation rate and minimum departure from nucleate boiling ratio (DNBR). The system will also monitor azimuthal flux tilt and total power level and will generate an alarm if any of these limits are exceeded. The margins to all of these limits except azimuthal tilt are continuously displayed to the operators; the tilt can be displayed at the request of the operator. The operator will monitor these margins and take corrective action if the limits are approached. These actions include improving the power distribution by moving full-length or part-length rods, reducing power, or changing thermal-hydraulic conditions, i.e., coolant inlet temperature and primary system pressure.

A description of the core operating limit supervisory system algorithms and an uncertainty analysis of the calculations performed by the core operating limit supervisory system is presented in CE Topical Report CENPD-169-P, "COLSS - Assessment of the Accuracy of PWR Operating Limits as Determined by the Core Operating Limit Supervisory Systems." We have reviewed this report and conclude that the methods employed in the core operating limit supervisory system to determine power distributions are acceptable.

Reactivity Coefficients

The reactivity coefficients are expressions of the effect on core reactivity of changes in such core conditions as power, fuel and moderator temperature, moderator density, and boron concentration. These coefficients vary with fuel burnup and power level. CE has presented calculated values of the coefficients in the FSAR and has also evaluated the accuracy of these calculations. We have reviewed the calculated values of reactivity coefficients and have concluded that they adequately represent the full range of expected values. We have reviewed the reactivity coefficients used in the transient and accident analyses and conclude that they conservatively bound the expected values, including uncertainties. Further, moderator and power Doppler coefficients along with boron worth are measured as part of the startup physics testing to assure that actual values are within those used in these analyses.

Control

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup and fission product buildup, a significant amount of excess reactivity will be built into the core. CE has provided sufficient information relating to core reactivity balance for the first core and has shown that means are incorporated into the design to control excess reactivity at all times.

Control of both excess reactivity and power level will be achieved with movable control element assemblies and through the variation of boron concentration in the reactor coolant. In addition, the chemical and volume control system will be capable of shutting down the reactor by adding soluble boron poison and maintaining the reactor at least five percent subcritical when refueling. The combination of control systems satisfies the requirement of GDC 26.

The CESSAR plants will be operated at steady-state full power with only one bank of the full-length control element assemblies slightly inserted. Limited insertion of the full-length control rods will permit compensating for fast reactivity changes (e.g., that required for power level changes and for the effects of minor variations in moderator temperature and boron concentrations) without impairing shutdown capability.

Rod insertion will be controlled by the power dependent insertion limits that will be given in the technical specifications. These limits will (1) ensure that there is sufficient negative reactivity available to permit the rapid shutdown of the reactor with ample margin, and (2) ensure that the worth of a control rod that might be ejected in the unlikely event of an ejected rod accident will be no worse than that assumed in the accident analysis.

Soluble boron poison will be used to compensate for slow reactivity change including those associated with fuel burnup, changes in xenon and samarium concentration, buildup of long-life fission products, burnable poison rod depletion, and the large moderator temperature change from cold shutdown to hot standby. The soluble boron poison system will provide the capability to take the reactor at least ten percent subcritical in the cold shutdown condition.

We have reviewed the calculated rod worths and the uncertainties in these worths, based upon appropriate comparison of calculations with experiments. On the basis of our review, we have concluded that the applicant's assessment of reactivity control is suitably conservative, and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability, assuming that the most reactive control element assembly is stuck in the fully withdrawn position. We have concluded that the control element assembly and soluble boron worths are acceptable for use in the accident analysis.

Stability

The stability of the reactor to xenon-induced power distribution oscillations and the control of such transients have been discussed in CESSAR. Due to the negative power coefficient, the reactor is inherently stable to oscillations in total reactor power.

The core may be unstable to axial xenon oscillations during the first cycle. The applicant has provided sufficient information to show that axial oscillations will be detected and controlled before any safety limits are reached, thus preventing any fuel damage. CE also concluded that the core will be stable to both radial and azimuthal xenon oscillations throughout core life based on analysis. This conclusion is verified by measurements on an operating reactor, Maine Yankee, in which the predicted damping factor agreed well with the measured value. We concur with this conclusion.

Vessel Irradiation

Maximum fast neutron fluxes greater than 1 Mev incident on the vessel and shroud inside diameters are presented. For reactor operation at the full power rating and an 80 percent capacity factor, the calculated vessel fluence greater than 1 Mev at the vessel wall does not exceed 3.15 x 10^{19} n/cm^2 over the 40-year design life of the vessel. The calculated exposure includes a 10 percent uncertainty factor. We conclude that acceptable values for the vessel fluence have been presented.

Criticality of Fuel Assemblies

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer and storage facilities. This design is the responsibility of the BOP designer for applications referencing CESSAR, as described in Section 9.1.

4.3.3 Analytical Methods

CESSAR describes the computer programs and calculational techniques used to calculate the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of these methods to predict experimental results. We conclude that the information presented adequately demonstrates the ability of these analytical methods to calculate the reactor physics characteristics of the CESSAR System 80 core.

4.3.4 Summary of Evaluation Findings

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission produce buildup, a significant amount of excess reactivity is designed into the core. CE has provided substantial information relating to core reactivity balances for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. CE has shown that sufficient control rod worth is available to shut down the reactor with at least a 2.0 percent $\delta k/k$ subcritical margin in the hot condition at any time during the cycle with the most reactive control rod stuck in the fully withdrawn position.

On the basis of our review, we conclude that the CESSAR assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control system to assure shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes available. We also conclude that nuclear design bases, features, and limits have been established in conformance with the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28.

CE has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of these methods to predict experimental results. The staff concludes that the information presented adequately demonstrates the ability of these analyses to predict reactivity and physics characteristics of the CESSAR design.

4.4 Thermal-Hydraulic Design

4.4.1 Thermal-Hydraulic Design Bases

The principal thermal-hydraulic design basis for the CESSAR core is the avoidance of thermal-hydraulic induced fuel damage during normal steady-state operation and anticipated operational transients. In order to satisfy the design basis, the digital core protection calculator provides for automatic trip or other corrective action to prevent violation of design limits. The design analysis is performed and design limits are established based on the criteria in the subsections which follow.

4.4.1.1 Departure From Nucleate Boiling

The margin to departure from nucleate boiling at any point in the core is expressed in terms of the departure from nucleate boiling ratio (DNBR). The DNBR is defined as the ratio of the heat flux required to produce departure from nucleate boiling at the calculated local conditions to the actual local heat flux.

The thermal-hydraulic design basis in the CESSAR FSAR Section 4.4.1.1 for the DNBR is as follows:

> "The minimum DNBR is such as to produce at least a 95% probability with 95% confidence that departure from nucleate (DNB) does not occur on a fuel rod having that minimum DNBR during steady-state operation and anticipated operational occurrences."

This design basis is evaluated in Section 4.2.

4.4.1.2 Hydrodynamic Stability

The hydrodynamic stability design basis in the CESSAR System 80 FSAR Section 4.4.1.2 is as follows:

> "Operating conditions do not lead to flow instability during steady-state operation or during anticipated operational occurrences."

We find this design basis acceptable, as described in Section 4.4.2.2.

4.4.1.3 Fuel Temperature

The fuel temperature basis is given in Section 4.4.1.3 of the FSAR as follows:

> "The peak temperature of the fuel is less than the melting point (5080°F unirradiated) and reduced by 58°F per 10,000 MWD/MTU during steady-state operation and anticipated operational occurrences."

This design basis is evaluated in Section 4.2

4.4.1.4 Core Flow

The minimum allowable reactor coolant flow less a maximum bypass flow (4.0 percent) is the design basis used in the thermal margin analysis. The minimum allowable reactor coolant flow is the total flow that the four reactor coolant pumps will produce at a 95% probability and 95% confidence level. This is a commonly used definition of core flow design basis and is acceptable.

4.4.2 Thermal-Hydraulic Design Metholodogy

4.4.2.1 Departure from Nucleate Boiling

The correlation used to determine the DNBR is the "CE-1" critical heat flux correlation. The CE-1 correlation is described in the CE Topical Report CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1 Uniform Axial Power Distribution." The report also describes the Combustion Engineering test program. The tests were conducted with 5 x 5 electrically heated rod bundles representative of 14 x 14 and 16 x 16 CE fuel assemblies.

The DNB program was extended by CE to include axially nonuniform heat flux data. The CE-1 critical heat flux correlation was modified to include the Tong F-Factor to account for the nonuniform heat flux. The test results and the modified form of the CE-1 correlation are documented in CENPD-207.

The staff has reviewed the CE-1 correlation based on the uniform and nonuniform axial heat flux test programs and concluded that the design limit DNBR based on the test data for 16 x 16 fuel should be 1.19 for fuel assemblies with standard grid configurations.

Since the CE-1 CHF correlation was based on test data having grid spacings of 14.3 inches, 17.4 inches, and 18.25 inches, the staff requested that CE justify using the CE-1 correlation for the CESSAR System 80 fuel design, which has one Inconel and ten Zircaloy grids per assembly, considering the specific number of grids and this spacing as compared to the test data. CE responded that the CESSAR System 80 fuel design has 15.7 inch grid spacing and that it is bounded by the 14.3 inch and 17.4 inch data. Lacking test data on the CESSAR System 80 bundle geometry, the staff is not convinced by arguments presented to justify the applicability of the available data but agrees that any effect on CHF of this spacer design change would be very small. Therefore, a 1 percent adjustment is imposed which results in a DNBR limit of 1.20.

Consequently, we require that the DNBR limit of 1.20 be used in the CPC software as a trip setpoint and that the DNBR value of 1.20 be included in the Technical Specifications.

4.4.2.2 Hydrodynamic Stability

In steady-state, two phase, heated flow in parallel channels, the potential for hydrodynamic instability exists. CE provided the following information in the FSAR to support the contention that the CESSAR System 80 core is thermal hydraulically stable. From literature, flow instabilities which have been observed occur almost exclusively in closed channel systems operating at low pressures relative to PWR pressures. For PWR operating pressures, experimental results have shown that, even with closed channel systems, operating limits due to the occurrence of critical heat flux are encountered before the flow stability threshold is reached. Kao, Morgan, and Parker conducted flow stability experiments at pressures up to 2200 psia with closed parallel heated channels. They found that at pressures above 1200 psia for flow and power levels encountered in power reactors no flow oscillations could be induced. It would be expected that the low resistance to coolant crossflow among subchannels of fuel assemblies would have a stabilizing effect, and that expectation is confirmed by experimental results by Veziroglu and Lee who found that crossflow between parallel heated channels enhances flow stability. Experimental evidence that flow instabilities will not occur is provided by the data from rod bundle DNB tests conducted by CE (CENPD-162-P-A and CENPD-207-P). Analytical support for the conclusion that flow instabilities will not reduce the thermal margin of CE PWRs is provided in a letter to J. F. Stolz, USNRC, from D. H. Williams, Arkansas Power & Light Co., January 16, 1978, and enclosure, "Assessment of Core Flow Stability for C-E PWRs," CEN-64(A)-P (proprietary) and CEN-64(A) (non-proprietary), July, 1977. This document presents an assessment of core flow stability for a typical C-E PWR using the CE-HYDNA code (Currin, H. B., et al., "HYDNA-Digital Computer Program for Hydrodynamic Transients in a Pressure Tube Reactor or a Closed Channel Core," Report CVNA-77, 1961). It was found that, for nominal coolant conditions, the flow is stable throughout the range of reactor power levels examined (100% - 250% rated power).

The staff is presently conducting a generic study of the hydrodynamic stability characteristics of pressurized water reactors. Limitations to the thermal-hydraulic design resulting from the staff study will be compensated by appropriate operating restrictions if necessary; however, no operating restrictions are anticipated. In the interim, the staff concludes that past operating experience, flow stability experiments, and the inherent thermal-hydraulic characteristics of CE pressurized water reactors provide a basis for accepting the CESSAR stability evaluation for issuance of an operating license.

4.4.3 Design Abnormalities

4.4.3.1 Fuel Rod Bowing

A significant parameter that influences the thermal-hydraulic design is rod-to-rod bowing within fuel assemblies. Presently, the staff is reviewing the CE Topical Report, CENPD-225, "Fuel and Poison Rod Bowing," which describes the methodology for evaluating the effects of rod-to-rod bowing in DNB. Until the staff completes its review of CENPD-225, we will impose a DNBR penalty which was calculated using the staff's interim criteria for evaluating the effects of rod bow on DNBR. Credit has been given for thermal margin due to a multiplier of 1.05 on the hot channel enthalpy rise factor used to account for pitch reduction due to manufacturing tolerances. The resultant reduction in DNBR due to rod bow is given by:

Burnup (MWD/MTU)	DNBR Penalty (%)
0	0
2,400	0
5,000	3.0
10,000	7.1
15,000	10.3
20,000	12.9
25,000	15.3
30,000	17.4
35,000	19.4
40,000	21.2

The appropriate DNBR penalty factors shall be included in the Core Operating Limits Supervisory System (COLSS) and CPC DNBR calculations.

The penalty for each batch will be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peak. A single net penalty for COLSS and CPC will be determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches. The appropriate provisions should be incorporated into the Technical Specifications. CE should also insert into the basis of the Technical Specifications any generic or plant specific margin that may be used to offset the reduction in DNBR due to rod bowing, and reference the source and staff approval of each generic margin. With these requirements satisfied by the reference plant applicants, the staff concludes that they have adequately accommodated the reductions listed above.

4.4.3.2 Crud Deposition

Crud deposition in the core and an associated change in core pressure drop and flow have been observed in some PWRs. In response to a staff question, CE stated that the effects of possible crud buildup have been accounted for in the CESSAR design in the form of an increase in the pressure drops used in the determination of design hydraulic loads. In addition, the core flow will be continuously monitored by the COLSS using pump casing differentials and pump speed as input. Any reduction in the core flow due to crud deposits will be accounted for in the COLSS thermal margin assessment.

Based on this information the staff concludes that CE has adequately addressed NRC concerns relative to uniform or preferential crud deposits in the core. NRC will require that the technical specifications include the requirements that the actual reactor coolant system total flow rate be greater than or equal to the value indicated by the core protection calculator system (CPCS).

4.4.4 Loose Parts Monitoring

The Loose Parts Monitoring System (LPMS) is not part of the CESSAR scope. The staff requires that the applicants referencing CESSAR provide the LPMS in conformance to the guidelines of Regulatory Guide 1.133.

4.4.5 Digital Core Protection Calculator

CE has indicated that the CESSAR and Arkansas Nuclear One, Unit 2 (ANO-2) Core Protection Calcaluators (CPCs) are basically identical. In addition to changes related to the Reactor Power Cutback System (RPCS), several software improvements will be implemented. The largest change in software from ANO-2 will be due to the RPCS. The RPCS will allow the NSSS to accommodate large turbine load rejections or the loss of one feedwater pump. CE estimates that RPCS will prevent, on the average, three plant trips per year. CE committed to provide the following:

(1) A CENPD document in February 1982 explaining the software difference from the San Onofre Units 2 and 3 CPC Functional Design Specification (CEN-147(S)-P) previously submitted on that docket. (The data base is plant specific and the initial submittal will be on the Arizona Public Service Company, Palo Verde 1, 2, and 3 docket, in August 1982.)

(2) A test plan with details of the test cases prior to the August 1982 submittal of the test report in order that the staff can comment on the appropriateness of the test cases. (A test report for verification of the software with the data base which is plant specific will be submitted by August 1982 under the Palo Verde 1 docket.)

The dates indicated above are not satisfactory. In order for the staff to make a timely review and not impact the startup schedule of Palo Verde Unit 1, which is currently scheduled to load fuel in November 1982, and also in the event that software changes are required as a result of the review, we require that the above information be submitted by March 1982.

We further require that CE clearly define the interface in responsibilities for the CPC system, with regard to plant-specific data base constants, software implementation testing, and the effects of the statistical combination of uncertainties, if used, on the DNBR limit.

We will require that the Technical Specifications include the requirements that the actual reactor coolant system flow rate be maintained at or above the minimum value used in the safety analysis and be consistent with that used for the core protection system.

4.4.6 Statistical Combination of Uncertainties (SCU)

CE has indicated that they may or may not apply the statistical combination of uncertainties (SCU) methods to combine DNBR uncertainties for CESSAR plants. IF SCU is used to combine DNBR uncertainties, the same methods as used for ANO-2 will be applied. We require that CE make a decision on the application of SCU for the System 80 plants by December 1981, together with submittals, in order to avoid possible startup delays due to inadequate review time.

4.4.7 Thermal-Hydraulic Models

The thermal-hydraulic design was performed using the TORC computer code. The TORC code, as described in CENPD-161, is used to analyze a specific three-dimensional power distribution superimposed on an explicit core inlet flow distribution. This type of calculation is performed in three stages. The first stage is to calculate the coolant conditions on a corewide basis. In the second stage, the hot assembly is analysed using input parameters obtained from the stage 1 calculations. Finally, the third stage is to calculate the local conditions and minimum DNBR by performing a subchannel analysis.

The design calculations for CESSAR were performed using the simplified TORC model discussed in CENPD-206-P. This method uses one limiting core radial power distribution for all the analyses; the hot assembly power is raised or lowered in order to obtain the proper maximum radial power factor; and all the assemblies except the hot assembly use the average mass velocity. Since the hot assembly can occur anywhere in the reactor core, a reduction in the amount of inlet flow to the hot assembly may be required. The percent reduction for mass velocity is determined by comparison of results with the detailed procedures discussed above.

The staff has previously reviewed CENPD-161 and CENPD-206 and determined that the TORC computer code and the simplified TORC model are acceptable methods for performing steady-state calculations of the reactor core thermal-hydraulic performance. The applications should be limited to conditions of single-phase flow or homogeneous two-phase flow.

4.4.8 Thermal-Hydraulic Comparison

The thermal-hydraulic design parameters for CESSAR are listed in Table 4.4-1. A comparison of these parameters with those of the Waterford Unit 3 and ANO-2 initial design parameters are also included. The major differences are increases in reactor core heat output, coolant flow rate, active heat transfer area and the nominal reactor inlet temperature. The system pressure, average velocity along the fuel rods, average temperature rise in the core and average heat flux are comparable. ANO-2 used the W-3 correlation compared to the CE-1 correlation used for Waterford 3 and CESSAR System 80. CESSAR uses thermal design procedures which are comparable to those of the Waterford 3 and San Onofre 2 initial designs.

4.4.9 N-1 Pump Operation

CE has submitted (in a letter dated October 5, 1981) an incomplete package for Appendix C of the CESSAR FSAR describing reactor operation with two reactor coolant pumps. The staff will require seven months for the review of the operation with two pumps after receiving the complete package from CE.

Until N-1 pump operation is approved we will not permit operation at power with less than four pumps operating or while in natural circulation. The staff will require that the Technical Specifications include appropriate provisions to ensure that these types of extended plant operation are prohibited until our review of this matter is complete.

4.4.10 Design Margin for Future Cycles

Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports, states that in Chapter 4 of the SAR:

"...the applicant should provide an evaluation and supporting information to establish the capability of the reactor to perform its safety functions throughout its design lifetime under all normal operation modes..."

The staff requested that CE demonstrate that there is sufficient thermal margin for operation in future cycles. CE responded that the CESSAR FSAR documents the ability of the core design to meet performance and safety requirements for the expected plant lifetime to the extent possible, based on information available prior to actual operation. Radial power distribution predictions as a function of burnup for the first four cycles are given. The rod radial power factor used in the DNB analyses of Section 4.4 of the FSAR is 1.55. The actual maximum rod radial power factor in the core will normally be lower but is not limited to a maximum value of 1.55. This value of 1.55 is higher than the prediction reported in Section 4.3 of the FSAR. CE indicated that the CPCs for CESSAR are basically identical to the design used in ANO-2. Also, CE stated that the application of statistical combination of uncertainties (SCU) that was used for ANO-2 is not presently planned for the first or future cycles of CESSAR plants. However, SCU may be applied for future cycles after receiving our approval.

Because of this weak assurance, the staff has reservations that the CESSAR thermal-hydraulic design, based on currently methodology, has margin to account for new fuel design and power distributions for all future cycles without incorporation of SCU. However, based on methodology revisions for ANO-2 (i.e., incorporation of statistical combination of uncertainties) we conclude that acceptable methodology changes can be incorporated, if necessary, to provide sufficient thermal margin for all future cycles.

We require that CE makes a decision in this regard by December 1981 with appropriate submittals if SCU is to be applied to Palo Verde 1.

4.4.11 Conclusions and Summary

The thermal-hydraulic design of the core for CESSAR has been reviewed. The scope of our review included the design basis and the steady-state analysis of the core thermal-hydraulic performance. The review concentrated on the difference between the proposed design and those designs that have been previously reviewed and found acceptable by the staff. All such differences were satisfactorily justified by CE. CESSAR's thermal-hydraulic analyses were preformed using analytical methods and correlations which have been previously approved by the Staff.

Based on our review and the design commitments provided by CE, the staff concludes that the thermal-hydraulic design of the CESSAR initial core conforms to the requirements of GDC 10, 10 CFR Part 50 and is acceptable for final design approval.

The reference plant applicant has overall responsibility for the startup test program. However, CESSAR defines CE's participation and provides guidelines to the applicant for a preoperational and initial startup test program in accordance with Regulatory Guide 1.68 to measure and confirm thermal-hydraulic design aspects. The staff has reviewed the CESSAR preoperational and initial startup test program and has concluded that it is acceptable, as described in Section 14.

The following open item is identified for future action:

Finalization and testing of the Core Protection Calculator (CPC) software system to include requirements of this SER:

(a) CE-1 limit change from 1.19 to 1.20 (without Statistical Combination of Uncertainties (SCU));

(b) appropriate rod bow compensation using results based on the approved staff model, and;

(c) the applicants commitment to supply the needed CPC information by March 1982, including the definition of interface responsibilities.

In order not to impact the startup schedule of Palo Verde Unit 1 and enable the staff to make a timely review, we require submission of the following information by the dates given:

(a) information on the Reactor Power Cutback System (RPCS) including the software report for the RPCS, a test report, including summary of the failure mode and transient analysis, by March 1982, with a test plan submitted sufficiently before that date to incorporate staff comments;

(b) the decision to incorporate SCU with applicable information by December 1981;

Information on operation with less than four reactor coolant pumps is to be supplied; the review will not be performed prior to the issue of the Palo Verde 1 operating license but may be applied to future cycles.

The following items need to be addressed in the Technical Specifications:

(a) the change in DNBR limit from 1.19 to 1.20;

(b) operating and surveillance requirements for the Loose Parts Monitoring System (LPMS) for the CESSAR-80 plants;

(c) requirements for the minimum core flow to be consistent with that used for the CPC;

(d) until information to be supplied for operation with less than four reactor coolant pumps is approved, the Technical Specifications should state that operation at power with less than four pumps operating or while in natural circulation is not permitted.

The following item is a correction to be made:

(a) Table 1.3-1 in the FSAR is to be amended to show the correct number of spacer grids as given in Section 4.4

Subject to resolution of the above items, we have concluded that the thermal-hydraulic design is acceptable.

TABLE 4.4-1 Reactor Design Comparison

	CESSAR	Waterford Unit-3	ANO-2
I. Performance Characteristics:			
Reactor Core Heat Output (MWt)	3,800	3,390	2,815
System Pressure, psia	2,250	2,250	2,250
Minimum DNBR at Steady-State Full Power	1.79	2.07	2.26
Minimum DNBR Limit	1.20	1.20	1.30
Critical Heat Flux Correlation	CE-1	CE-1	W-3
II. Coolant Flow:			
Total Flow Rate (gpm)	446,000	396,000	322,000
Effective Flow Rate for Heat Transfer (gpm)	428,000	382,000	310,700
Average Velocity Along Fuel Rods, (ft/s)	16.4	16.3	16.4
Average Mass Velocity (10^6) $lb/hr\text{-}ft^2$)	2.58	2.61	2.6
III. Coolant Temperature, °F:			
Nominal Reactor Inlet	565	553	553.5
Average Rise in Core	59	60	58.5
IV. Heat Transfer, 100 Percent Power:			
Active Heat Transfer Surface Area, (ft^2)	68,600	62,000	51,000
Average Heat Flux ($BTU/hr\text{-}ft^2$)	184,400	182,400	182,200
Maximum Allowable Heat Flux ($BTU/hr\text{-}ft^2$)	478,074	457,708	494,738
Average Linear Heat Rate (kw/ft)	5.4	5.34	5.34
Peak Allowable Linear Heat Generation Rate (kw/ft)	14.0	13.4	14.5

4.5 Reactor Materials

4.5.1 Control Rod Drive Structural Materials

We conclude from our review that the materials used for the construction of the Control Rod Drive Structure Materials are acceptable and meet the applicable portions of the requirements of GDC 1, 14, and 26 of Appendix A and Section 50.55a of 10 CFR Part 50.

CESSAR has met the requirements of GDC 1, 14, and 26 and Section 50.55a of 10 CFR Part 50 by assuring that the design, fabrication, and testing of the materials used in the Control Rod Drive Structural Materials are of high quality standards and adequate for structural integrity.

The mechanical properties of structural materials selected by CE for the control rod system components that are exposed to the reactor coolant designer and conform with our position as stated in Section 4.5.1 of the Standard Review Plan that the yield strength of cold worked austenitic stainless steel should not exceed 90,000 pounds per square inch.

The controls imposed upon the austenitic stainless steel of the mechanisms conform to the recommendations of Regulatory Guide 1.31, "Controls of Ferrite Content in Stainless Steel Weld Metal," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices performed in accordance with these recommendations provide assurance that the incidents of intergranular stress corrosion cracking will be minimized during the design life of the components. The compatibility of the materials used in the control rod system, in contact with the reactor coolant, satisfy the criterion of Subarticle NB-2120 of Section III of the Code. Both martensitic and precipitation hardened stainless steel have been tempered in accordance with our position. Justification for the use of ASTM A276 Type 440C was provided in the CESSAR FSAR. The yield strength of the materials exceed 90 Ksi, but corrosion has not occurred in operating reactors.

Cleaning and cleanliness control have been performed to provide adequate contamination control of components during fabrication, shipment and storage.

The controls imposed upon the austenitic stainless steel of the system will satisfy the requirements of the materials specification. Most of the austenitic stainless steel materials are furnished in solution heat treated condition. Sensitization is avoided by not permitting heat treatment in the temperature range of 800°F to 1500°F. The fabrication and heat treatment practices provide assurance that intergranular stress corrosion cracking will be minimized during the design life of the components.

Conformance with the codes, standards and regulatory guides, conformance with our positions on the allowable maximum yield strength of cold worked austenitic stainless steel, and generally the tempering or aging temperatures of martensitic and precipitation hardened stainless steel, constitute an acceptable basis for meeting the requirements of GDC 1, 14 and 26 and Appendix A and Section 50.55a of 10 CFR Part 50.

4.5.2 Reactor Internals and Core Support Materials

We conclude from our review that the materials used for the construction of the reactor internals and core support are acceptable and meet the applicable portions of the requirements of GDC 1 of Appendix A and Section 50.55a of 10 CFR Part 50.

CE has met the requirements of GDC 1 and Section 50.55a of 10 CFR Part 50 by assuring that the design, fabrication, and testing of the materials used in the reactor internals and core support structure are of high quality standard and adequate for structural integrity. The controls imposed upon components constructed of austenitic stainless steel satisfy the intent of the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content of Stainless Steel Weld Metal," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

The materials used for the construction of components of the reactor internals and core support structure have been identified by specification and found to be in conformance with the requirements of Article NG-2000 of Section III and Parts A, B, and C of Section II (or equivalent specification) of the ASME Code. As proven by extension test and satisfactory performance the specified materials are compatible with the expected environment and corrosion is expected to be negligible.

The controls imposed on the reactor coolant chemistry will provide reasonable assurance that the reactor internals and core support structure will be adequately protected during operation from conditions which could lead to stress corrosion of the materials and loss of component integrity.

The material selection, fabrication practices, examination and testing procedures, and control practice performed in accordance to these recommendations will provide reasonable assurance that the materials used for the reactor internals and core support structure will be in a metallurgical condition to preclude service deterioration. Conformance with requirements of the ASME Code and the recommendations of the Regulatory Guides constitute an acceptable basis for meeting the requirements of GDC 1 and Section 50.55a of 10 CFR Part 50.

4.6 Functional Design of Reactivity Control Systems

The functional design of the reactivity control systems for the facility is within the CESSAR scope and has been reviewed by the staff to confirm that they meet the various reactivity control conditions for all modes of operation. These are:

(1) The capability to operate in the unrodded, critical, full power mode throughout plant life.

(2) The capability to vary power level from full power to hot shutdown and assure control of power distributions within acceptable limits at any power level.

(3) The capability to shutdown the reactor in a manner sufficient to mitigate the effects of postulated events discussed in Section 15.0 of this SER.

The reactivity control systems for the facility are the control element assembly drive system (CEADS), the safety injection system (SIS) and the chemical and volume control system (CVCS).

The CEADS contains magnetic jack control element drive mechanisms (CEDM). When electrical power is removed from the coils of the CEDM, the armature springs automatically cause the driving and holding latches to be withdrawn from the CEDM drive shaft, allowing insertion of the control element assemblies (CEA) and the part length control element assemblies (PLCEA) by gravity. There are 76 full length CEA and 13 PLCEA. The regulating CEA groups (full- and part-length) may be used to compensate for changes in reactivity associated with power level changes and power distribution, variations in moderator temperature, or changes in boron concentration. Refer to Sections 3.9.4 and 4.3 of this SER for further discussion of this feature. No reactivity credit toward shutdown margin is taken for the PLCEA. The PLCEA contain a strong neutron absorber in the top 10 percent of their active legnth, which on reactor trip offsets any positve reactivity insertion due to the shift in axial flux distribution between full and zero power. They also help control power distribution and suppress xenon-induced power oscillations.

The SIS is automatically actuated to inject borated water into the reactor coolant system (RCS) upon receipt of a safety injection actuation signal (SIAS). The SIS pumps take suction from the refueling water storage tank (RWST). The SIS is discussed further in Section 6.3 of this SER.

The CVCS is designed to control slow or long-term reactivity changes such as that caused by variation in coolant temperature, fuel burnup, or variations in the xenon concentration. The CVCS controls reactivity by adjusting the dissolved boron concentration in the reactor coolant system. The boron concentration is controlled to obtain optimum CEA positioning, to compensate for reactivity changes during startup, load following (changes in reactor power level), shutdown, and to provide shutdown margin for maintenance and refueling operations or emergencies. The boric acid concentration in the reactor coolant system is controlled by the charging and letdown portions of the CVCS.

The CVCS can be used to maintain reactivity within the required bounds by means of the automatic makeup system which replaces minor coolant leakage without significantly changing the boron concentration in the reactor coolant system. Dilution of the reactor coolant system boron concentration is required to compensate for the reactivity losses occurring as a result of fuel and burnable poison depletion. This is accomplished by manual operation of the CVCS. The CVCS is discussed further in Section 9.3.4 of this SER.

The concentration of boron in the reactor coolant system is changed manually under the following operating conditions:

(1) Startup - boron concentration decreased to compensate for moderator temperature and power increase.

(2) Load follow - boron concentration increased or decreased to compensate for xenon transients following load changes.

(3) Fuel burnup - boron concentration decreased to compensate for burnup.

(4) Cold shutdown - boron concentration increased to compensate for increased moderator density due to cooldown.

Soluble poison concentration is used to control slow operating reactivity changes. If necessary, CEA movement can also be used to accommodate such changes, but assembly insertion is used mainly to mitigate anticipated operational occurrences (the analysis assumes a single malfunction such as a stuck rod). In either case, fuel design limits will not be exceeded. The soluble poison control is capable of maintaining the core subcritical under conditions of cold shutdown which conforms to the requirements of GDC 26, "Reactivity Control System Redundancy and Capability."

Full-length CEA are the primary shutdown mechanism for normal operation, accidents, and transients. They are inserted automatically in accident and transient conditions. Concentrated boric acid solution is injected by the SIS in the event of a loss-of-coolant accident, steam line break, loss of normal feedwater flow, steam generator tube rupture, or CEA ejection, thereby complying with GDC 20, "Protection System Functions," which requires that automatic protective systems be provided (1) to assure that specified acceptable fuel design limits are not exceeded and (2) to sense accident conditions and to initiate operation of systems and components important to safety.

The ability of the full-length CEA to be fully inserted and the PLCEA to have their position changed is tested every 31 days during power operation. At every refueling shutdown each CEA is stepped over its entire range of movement, and drop tests are performed to demonstrate the ability of the assemblies to meet required drop times. The CEA design is such that a single failure will not result in loss of the protection system nor will a loss of redundancy occur as a result of removal of a channel or component from service. This is discussed further in Section 7.2 of this SER. The foregoing periodic testing, reliability, and redundancy conform to the requirements of GDC 21, "Protection System Reliability and Testability."

Failure of electrical power to any control element drive mechanism will result in insertion of that assembly. Analysis of accidental withdrawal of a CEA was found to have acceptable results as discussed in Section 15.2.4 of this SER. This conforms to the requirements of GDC 23, "Protection System Failure Modes," and 25, "Protection System Requirements for Reactivity Control Malfunctions."

The reactivity control systems, including the addition of concentrated boric acid solution by the safety injection system, are capable of controlling all anticipated operational changes, transients, and accidents. All accidents are calculated with the assumption that the most reactive CEA is stuck and cannot be inserted, which complies with the requirements of GDC 27, "Combined Reactivity Control Systems Capability." Compliance with the requirements of GDC 28, "Reactivity Limits," is discussed in Sections 4.3 and 15.2 of this SER.

Based on our review, we conclude that the reactivity control system functional design meets the requirements of GDC 21, 23, 25, 26, and 27 with respect to its reliability and testability, fail safe design, malfunction protection design, redundancy and capability, and combined systems capability and is, therefore, acceptable.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

The CESSAR reactor design is a pressurized water reactor (PWR) with two coolant loops. The reactor coolant system (RCS) circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to a secondary (steam generating) system. In a PWR, the steam generators provided the interface between the reactor coolant (primary) system and the main steam (secondary) system. The steam generators are vertical U-tube heat exchangers, with an integral economizer section, in which heat is transferred from the reactor coolant to the main steam system. Reactor coolant is prevented from mixing with the secondary steam by the steam generator tubes and steam generator tube sheet, making the RCS a closed system and, thus, forming a barrier to the release of radioactive materials from the reactor core to the containment building. The majority of the secondary system, except for interface requirements, is outside the scope of CESSAR.

The major components of the RCS are the reactor vessel; two parallel heat transfer loops, each containing one steam generator and two reactor coolant pumps; a pressurizer connected to one of the reactor vessel outlet pipes; and the associated reactor coolant piping.

The pressure in the primary system is controlled by the pressurizer, where steam and water are maintained in thermal equilibrium. Either steam is formed by energizing immersion heaters in the pressurizer or steam is condensed by the pressurizer sprays to limit the pressure variation caused by contraction or expansion of the reactor coolant. The average temperature of the reactor coolant varies with power level and the fluid expands or contracts, changing the pressurizer water level.

The major reactor coolant loop penetrations are the pressurizer surge line in one of the reactor vessel outlet pipes; the four safety injection inlet nozzles, one in each reactor vessel inlet pipe; one outlet nozzle to the shutdown cooling system (SDCS) in one reactor vessel outlet pipe; two pressurizer spray nozzles; vent and drain connections; and sample and instrument connections.

Overpressure protection for the reactor coolant pressure boundary (RCPB) is provided by four spring-loaded ASME Code safety valves connected to the top of the pressurizer. These valves discharge to the reactor drain tank, where the steam is released under water to be condensed and cooled. If the steam discharge exceeds the capacity of the reactor drain tank, it is relieved to the containment atmosphere via a rupture disc installed in the tank.

The interface requirements on the reference plant design, which assure the functional performance of the reactor coolant system, are specified in Section 5.1.4 of the CESSAR FSAR. We have reviewed these interface requirements in the context of the Palo Verde reference plant application and found them acceptable. However, we have not, as yet, completed our review with regard to the acceptability and completeness of the interface requirements for future plant applications. We will report on the resolution of this issue in a revision to this report.

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR Part 50, Section 50.55a

The components of the reactor coolant pressure boundary (RCPB), as defined by the rules of 10 CFR Part 50, Section 50.55a, "Codes and Standards," have been properly classified in Table 5.2-1 of CESSAR as American Society of Mechanical Engineers (ASME) Section III, Class 1 components. These components are designated Safety Class 1 (Quality Group A) in conformance with Regulatory Guide 1.26 in Table 3.2-1 of the CESSAR FSAR. However, Table 5.2-1 of the CESSAR FSAR does not identify the applicable ASME Code Edition and Addenda used in the construction of the RCPB components. Therefore, in order to assure compliance with Section 50.55a of 10 CFR 50, we will require that each application referencing CESSAR identify the ASME Code Edition and Addenda used in the construction of the reference plant ASME Section III, Class 1 components of the RCPB.

In addition to the Quality Group A components of the RCPB, certain lines that perform a safety function and which meet the exclusion requirements of footnote 2 of the rule are classified Safety Class 2 (Quality Group B) in accordance with the guidance provided in Regulatory Position C.1 of Regulatory Guide 1.26 and are constructed as ASME Section III, Class 2 components. We conclude that construction of the components of the reactor coolant pressure boundary in conformance with the appropriate ASME Code Editions and Addenda and the Commission's regulations provides assurance that component quality is commensurate with the importance of the safety function of the reactor coolant pressure boundary and constitutes an acceptable basis for satisfying the requirements of GDC 1 and is, therefore, acceptable.

5.2.1.2 Applicable Code Cases

We will review, in the Safety Analysis Report that references CESSAR, those specific Code Cases of the American Society of Mechanical Engineers that are applied in the construction of ASME Section III, Class 1 components within the reactor coolant pressure boundary (Quality Group A). Only those Code Cases found to be acceptable in Regulatory Guide 1.84, "Code Case Acceptability - ASME Section III, Design and Fabrication," and Regulatory Guide 1.85, "Code Case Acceptability - ASME Section III, Materials," may be used in the construction of reactor coolant pressure boundary components.

We conclude that compliance with the requirements of these Code Cases will result in a component quality level that is commensurate with the importance of the safety function of the reactor coolant pressure boundary and constitutes an acceptable basis for satisfying the requirements of General Design Criterion 1 and is, therefore, acceptable.

5.2.2 Overpressurization Protection

Overpressure protection of the primary coolant system is designed to accommodate both low and high temperature operation. High temperature overpressure protection is designed to limit transient pressures to below 110% of design pressure. Low temperature overpressure protection is designed to prevent the RCS from exceeding 10 CFR Part 50, Appendix G, "Fracture Toughness," limits.

The interface requirements on the BOP for overpressure protection are specified in Section 5.1.4 of the CESSAR FSAR. We conclude that these interface requirements are acceptable and complete on the following basis: (1) the interface requirement specifies an adequate relief capacity for each primary safety valve such that the total relief capacity is sufficient to maintain the primary coolant pressure below the maximum allowable pressure during any transient or accident and, therefore, conform to GDC 15; and (2) the interface requirements specifies an adequate relief capacity for each secondary safety valve such that the total secondary relief capacity is sufficient to limit the maximum steam generator pressure to 110% of the system design pressure in accordance with the requirements of the ASME Code.

5.2.2.1 High Temperature Overpressure Protection

The high temperature overpressure protection design basis is to maintain secondary and primary operating pressures within 110% of design by means of four primary safety valves, 20 secondary safety valves, and the reactor protection system. The secondary safety valves are sized to pass a total steam flow of 19×10^6 pounds per hour at 1000 psia which is greater than the rated steam flow at full power of 17.2×10^6 pounds per hour at 1070 psia. The reactor is designed to trip at an RCS pressure of 2450 psia while the primary pressurizer safety valves are designed to lift at a pressure of 2500 psia, which is the RCS design pressure.

The design basis event for evaluating sizing adequacy of the primary safety valves is a loss of load with a delayed reactor trip. In the analysis provided, no credit is taken for letdown, charging, pressurizer spray, secondary bypass, or feedwater flow after turbine trip. At the onset of the transient, the RCS and main steam supply system (MSSS) are at the maximum rated output plus a 2% uncertainty. The moderator and Doppler coefficients used for the analysis maximize the pressure and power excursion.

The CESSAR design includes a supplementary protective system (SPS) in addition to the reactor protection system (RPS). The SPS provides a completely diverse high pressure trip at the same setpoint as the RPS. An upper limit on instrument error is used in CE analysis. This delays the reactor trip, which in turn provides additional conservatism in assessing the sizing adequacy of the RCS safety valves. Also, this design meets acceptance criteria II.2.C of SRP Section 5.2.2, which states that the reactor trip should be initiated either by the high pressure signal or by the second signal from the RPS, whichever is later.

Under the assumptions of this analysis, the peak primary and secondary system pressure are limited to 110% of design pressure during the loss of load transient.

Testing and inspection of the primary safety valves is governed by the ASME Code, Section XI, Subsection IWV. The secondary safety valves will be tested to verify setpoints during hot functional testing. Periodic inservice testing of the secondary valves will be defined in the Technical Specifications.

We have concluded that the high temperature overpressure protection system meets the staff criteria as specified in SRP Section 5.2.2 and is, therefore, acceptable.

5.2.2.2 Low Temperature Overpressure Protection

Overpressure protection of the RCS during low temperature conditions is provided by relief valves SI-179 and SI-189 located in the shutdown cooling system (SDCS) suction lines. An SDCS relief valve is a spring-loaded (bellows) liquid relief valve with a capacity of 4,000 gal/min at 450 psia with 10% accumulation. The most limiting transients calculated were an inadvertent safety injection actuation (mass input) and an reactor coolant pump start when a positive steam generator to reactor vessel ΔT exists (energy input). Calculations show that this relief system can mitigate these transients and prevent violation of 10 CFR Part 50, Appendix G, limits.

The SDCS relief valves are sized based on an inadvertent safety injection actuation signal (SIAS) with full pressurizer heaters operating from a water-solid condition. The SIAS assumes simultaneous operation of two HPSI pumps and three charging pumps with letdown isolated. The resulting flow capacity requirement for water is 4000 gpm.

The valve supplied by the valve manufacturer has a rated relief capacity of 5180 gpm. This rated relief capacity exceeds the required relief flow for the worst transient with sufficient margin in relieving capacity.

Alignment of the SDCS relief valve to the RCS is provided via plant procedures to ensure RCS overpressure protection for all temperatures below the pressure-temperature (P-T) operating curve limits, corresponding to the pressurizer safety valve set pressure of 2500 psia. The requirements for alarms to ensure that the system is aligned to the LTOP mode and to indicate transients in progress are discussed in Section 7.1 of this report.

The staff will require that the plant Technical Specifications include the following:

(1) A Technical Specification shall be imposed to ensure the RCS is on the SDCS with all suction line valves open whenever the RCS temperature is below an appropriate temperature and the RCS pressure is below the SDCS design pressure.

(2) A Technical Specification to prohibit actuation of a reactor coolant pump if the associated steam generator to RCS ΔT is greater than 150°F.

(3) The setpoint for automatic isolation of the SDCS will be raised to 700 psig.

(4) A Technical Specification to test the SDCS safety relief valves at intervals not to exceed 30 months shall be imposed.

System design criteria required by the staff include no credit for operator action for 10 minutes; the mitigating system must meet single active failure criteria; the system must be testable; the system must be able to withstand an operating basis earthquake (OBE); and the system must be capable of functioning following loss of offsite power. CESSAR has met all the design criteria for the NRC's position on water solid overpressure protection. This provides adequate assurance that the temperature/pressure limits presented in Appendix G of 10

CFR Part 50 will not be exceeded during any anticipated operational occurrence and postulated accidents. The system proposed is, therefore, acceptable.

5.2.3 Reactor Coolant Pressure Boundary Materials

Reviews of the materials for the reactor vessel and for the steam generators (primary side) are given in Sections 5.3.1 and 5.4.2.1, respectively, of this SER.

The staff concludes that CESSAR is acceptable and the reactor coolant pressure boundary (RCPB) materials meet the requirements of General Design Criteria 1, 4, 14, and 30 of Appendix A of 10 CFR Part 50; the requirements of Appendix B of 10 CFR Part 50; and the requirements of §50.55 of 10 CFR Part 50. This conclusion is based on the staff's review of CESSAR, covering all of the RCPB materials in the CESSAR scope. We will review RCPB materials in the BOP scope in each application referencing CESSAR.

5.2.3.1 Material Specifications and Compatibility with Reactor Coolant

The materials used for construction of components of the reactor coolant pressure boundary (RCPB) have been identified in CESSAR by specification and found to be in conformance with the requirements of Section III of the ASME Code. Compliance with the above Code provisions for materials specifications satisfies the quality standards requirements of GDC 1 and 30 and §50.55.

The materials of construction of the RCPB exposed to the reactor coolant have been identified and all of the materials are compatible with the primary coolant water, which will be chemically controlled in accordance with appropriate technical specifications. This compatibility has been proven by extensive testing and satisfactory performance. This includes conformance with the recommendations of Regulatory Guide 1.44, "Control of Sensitized Stainless Steel," regarding prevention of significant sensitization and surface contamination of austenitic stainless steel. General corrosion of all materials will be negligible. All of the ferritic steels that are part of the RCPB are provided with corrosion resistant cladding on all surfaces that are exposed to the reactor coolant. The above evidence of compatibility with the coolant and compliance with the regulatory guide recommendations satisfy the requirements of GDC 4 relative to compatibility of components with environmental conditions.

The materials of construction for the RCPB are compatible with the thermal insulation used in these areas. Two types of insulation will be used: stainless steel reflective and nonmetallic. The nonmetallic thermal insulation materials that are used are in accordance with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." Conformance with the above recommendations satisfies the requirements of GDC 14 relative to prevention of leakage and failure of the RCPB.

5.2.3.2 Fabrication and Processing of Ferritic Materials

Fracture toughness of the components of ferritic materials in the reactor coolant boundary is covered in Section 5.3.1 of this SER. All ferritic steel tubular products used for components of the RCPB will be found acceptable by nondestructive examination in accordance with the requirements of the applicable editions and addenda of the ASME Code, Section III, by which the products were procured. Compliance with these Code requirements satisfies the quality standards requirements of General Design Criteria 1 and 30 and §50.55a.

Welding of all components of ferritic steels in the RCPB will be performed in accordance with the provisions of the ASME Code, Sections III and IX. This compliance with the Code provides reasonable assurance that cracking of components made from ferritic steels will not occur during fabrication. The electroslag welding process will not be used in the fabrication of RCPB components of ferritic steels.

The controls imposed on welding preheat temperatures for welding ferritic steels satisfy the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low-Alloy Steels." Although Position 2 of the guide will not be complied with, this will be compensated for by verifying the soundness of all welds by ASME acceptable examination procedures, in accordance with the Position 4 recommendation. The controls imposed provide reasonable assurance that cracking of components made from low alloy-steels will not occur during fabrication and minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment. These controls satisfy the quality standards requirements of General Design Criteria 1 and 30 and §50.55a.

The controls imposed during weld cladding low-alloy steel components with austenitic stainless steel conform with the major recommendations of Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." The controls consist of limited use of SA-508, Class 2, forging material and avoidance of high-heat-input weld cladding processes. These controls provide assurance that practices that could result in underclad cracking will be restricted. The controls also satisfy the quality standards requirements of General Design Criteria 1 and 30 and §50.55a.

Performance qualifications for personnel welding components of ferritic steels under conditions of limited accessibility are conducted and maintained in accordance with the requirements of Sections III and IX of the ASME Code. A requalification will be required when any of the essential variables of Section IX are changed or when authorized personnel have a reason to question the ability of the welder to perform satisfactorily to the applicable requirements. Production welding will be monitored for compliance with the procedure parameters, welding qualification requirements were certified in accordance with Sections III and IX, and weld quality was verified by performance of the required nondestructive examination. The controls imposed on welding ferritic steels under conditions of limited accessibility provide assurance that proper requalification of welders will be required in accordance with welding conditions and that welds of ferritic steel components in the RCPB will be satisfactory in locations of restricted accessibility. These controls also satisfy the quality standard requirements of General Design Criteria 1 and 50 and §50.55a.

5.2.3.3 Fabrication and Processing of Austenitic Stainless Steel

All austenitic stainless steel tubular products of the RCPB in the CESSAR scope will be acceptably tested by nondestructive examinations in accordance with the requirements of the applicable editions and addenda of the ASME Code, Section III, by which the products were procured. Compliance with these Code requirements satisfies the quality standards requirements of GDC 1 and 30 and §50.55a.

No austenitic stainless steel components of the RCPB will have yield strength exceeding 90,000 psi, in accordance with our position as stated in Section 5.2.3 of the Standard Review Plan.

The controls during material procurement, fabrication, shipment, and storage of the NSSS components of austenitic stainless steel to avoid stress corrosion cracking by avoiding sensitization and surface contaminants will be in conformance with all of the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

The specific requirements for cleanliness and contamination protection in the equipment specifications for NSSS components fabricated from austenitic stainless steel will be in accordance with the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," except for halide content of the cleaning water (<0.60 ppm chloride and <0.40 ppm fluoride). These contents are a little higher than those of reactor grade water, recommended for final flushing by Regulatory Guide 1.37, but are sufficiently low, combined with proper packaging to protect from weather, dirt, wind, water spray, and detrimental environmental conditions encountered during shipment and subsequent site storage, to ensure freedom from detrimental surface contamination. The flushing water is also inhibited by the addition of hydrazine.

The controls followed in order to prevent excessive yield strength, sensitization, and surface contamination, described in the preceding paragraphs, provide reasonable assurance that the RCPB components of austenitic stainless steels will be in a metallurgical condition that minimized susceptibility to stress corrosion cracking during service. These controls meet the requirements of GDC 41 relative to compatibility of components with environmental conditions, the requirements of GDC 14 relative to prevention of leakage and failure of the RCPB, and the requirements of Appendix B relative to cleaning of material to prevent deterioration.

The controls imposed during welding of RCPB components of austenitic stainless steel will be in accordance with the requirements of the ASME Code, Section III and IX. No electroslag welding will be used. In order to preclude microfissuring in austenitic stainless steel welds in NSSS components in the RCPB, the major recommendation of the Interim Position in effect for Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," will be followed. Conformance with the major recommendation consists of controlling weld filler metal to deposit a minimum of 3% delta ferrite. The controls imposed during welding of austenitic stainless steel components in the RCPB will provide reasonable assurance that no deleterious hot cracking will be present during the fabrication and assembly of these components. The conrols meet the requirements of GDC 14 relative to prevention of leakage and failure of the RCPB.

For NSSS components, welder qualification for welding austenitic stainless steel components under conditions of limited accessibility will be the same as for welding ferritic steel components under these conditions, as described above.

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

The CESSAR design has incorporated ready access and other provisions for routine inservice inspection and testing. The inspection and testing program, however, are the responsibility of the applicants referencing CESSAR. We will evaluate the RCPB inservice inspection and testing program in each reference plant application.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

A limited amount of leakage is to be expected from components forming the reactor coolant pressure boundary (RCPB). Means are provided for detecting and identifying this leakage in accordance with the requirements of General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary." Leakage is classified into two types, identified and unidentified. Components such as valve stem packing, pump shaft seals, and flanges are not completely leaktight. Since this leakage is expected, it is considered identified leakage and is monitored, limited, and separated from other leakage (unidentified) by directing it to closed systems as identified in the guidelines of Position C.1 of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

CESSAR indicates that the design of systems for detection of unidentified leakage is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Section 5.1.4 concerning sources, disposition, and indication of unidentified reactor coolant pressure boundary leakage and intersystem leakage for those areas within the CESSAR scope in accordance with the requirements of GDC 30 and the guidelines of Regulatory Guide 1.45.

Sources, disposition, and indication of identified leakage within the CE scope are as follows:

(1) Abnormal reactor coolant pump seal leakage into containment is detected by the seal pressure indicators and the reactor drain tank level indicators. Seal leakage through the reactor coolant pump seal cooler to the component cooling water system is detected by the component cooling water temperature, surge tank level, and radiation monitors. All the above indications with associated alarms are provided in the control room.

(2) Pressurizer safety valve leakage to the reactor drain tank is monitored in the main control room by the temperature indicators and alarms on the pressurizer safety valve discharge line and the level, temperature, and pressure indicators and alarms on the reactor drain tank.

The means for detecting steam generator tube or tube sheet leakage, although an unidentified source, is included in CESSAR. An increase in the radioactivity indicated by the monitors in the condenser air removal system and steam generator blowdown system as well as routine steam generator water samples would indicate reactor coolant pressure boundary leakage into the secondary system.

Based on our review of CESSAR, we conclude that the means provided for detection of identified reactor coolant pressure boundary leakage are in accordance with the requirements of GDC 30 and the guidelines of Regulatory Guide 1.45 and are, therefore, acceptable. We further conclude that the interfaces identified in CESSAR provide adequate information relating to unidentified leakage in order that referencing applicants can comply with the requirements of GDC 30 and the guidelines of Regulatory Guide 1.45 and are, therefore, acceptable and complete in this regard.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

The staff concludes that the reactor vessel materials will be acceptable and that the processes applied to these materials for fabrication and inspection are acceptable and meet the requirements of GDC 1, 14, 30, 31, and 32 of Appendix A, Appendix B of 10 CFR Part 50, and §50.55a. This conclusion is based on the following considerations.

The materials to be used for construction of the reactor vessel and the appurtenances within the CESSAR scope have been identified by specification and found to be in conformance with Section III of the ASME Code. Special requirements of CE with regard to control of residual elements in ferritic materials have been identified and are considered acceptable.

Special processes used for manufacture or fabrication of the reactor vessel and its appurtenances have been identified, and appropriate data reports on each process as required by Section III of the ASME Code will be submitted by the reference plant applicant. Since certification has been made by CE that the materials and fabrication requirements of Section III of the Code have been complied with, the special processes to be used are considered acceptable.

Special methods to be used for nondestructive examination of the reactor vessel and its appurtenances have been identified and have been found equivalent or superior to the techniques described in Appendix X of the ASME Code, Section III. Demonstrations will be made using these special techniques, and will satisfy all requirements of the Code. The special methods of nondestructive examination are deemed acceptable.

Special controls and special welding processes used for welding the reactor vessel and its appurtenances have been identified and will be qualified in accordance with the requirements of Code Sections III and IX. The controls imposed on welding preheat temperatures will be in conformance with the requirements of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels," and provide reasonable reassurance that cracking of components made from low alloy steels will not occur during fabrication, and will minimize the possibility of subsequent cracking due to residual stresses being retianed in the weldment. The controls imposed upon austenitic stainless steel welds are in conformance with Regulatory Guide 1.31, "Control of Stainless Steel Welding."

GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A, 10 CFR Part 50, requires that the reactor coolant pressure boundary be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and anticipated transient conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. GDC 32, "Inspection of Reactor Coolant Pressure Boundary," of Appendix A, 10 CFR Part 50, requires, in part, that the reactor coolant pressure boundary be designed to permit an appropriate material surveillance program for the reactor pressure vessel.

We have reviewed the materials selection, toughness requirements, and extent of materials testing conducted by CE in accordance with the above criteria subject to the rules and requirements of 10 CFR Part 50 Paragraph 50.55a "Codes and

Standards;" 10 CFR Part 50 Appendix G, "Fracture Toughness Requirements;" and 10 CFR Part 50 Appendix H, "Reactor Vessel Materials Surveillance Program Requirements."

The Edition and Addenda of the ASME Code which are applicable to the design and fabrication of any reactor vessel are specified in Section 50.55a of 10 CFR Part 50 and are based on the construction permit date. CESSAR indicates that construction permits for all System 80 plants have or will be issued on or after May 25, 1976. Based on the CESSAR current reference plant construction permits dates, Paragraph 50.55a of 10 CFR Part 50 requires that reactor coolant pressure boundary materials meet the fracture toughness requirements of at least the 1971 Edition of the ASME Code, Summer 1972 Addenda. CESSAR indicates all System 80 nuclear plants will be fracture toughness tested to the requirements of Paragraph 50.55a of 10 CFR Part 50. Therefore, all reactor coolant pressure boundary materials in CESSAR nuclear plants must be fracture toughness tested to at least the 1971 Edition of the ASME Code, Summer 1972 Addenda. As stated in Section 5.2.1.1 of this report, each reference plant will be required to identify the applicable ASME Code Edition and Addenda.

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Requirements," of 10 CFR Part 50 specify the fracture toughness requirements for the ferritic materials of the reactor coolant pressure boundary.

<u>Compliance to Appendix G, 10 CFR Part 50</u>

CESSAR indicates System 80 plants will comply with all Appendix G, 10 CFR Part 50 requirements except for Sections III.B.5 and III.C.2. Our evaluation of the CE request for exception to Sections III.B.5 and III.C.2 of Appendix G, 10 CFR Part 50 follows.

Section III.B.5 of Appendix G, 10 CFR Part 50, requires that the fracture toughness test results include a certification that the tests were performed in compliance with the requirements of Appendix G, 10 CFR Part 50. CESSAR indicates that the test reports will be certified to conform with the requirements of the applicable ASME Code Edition and Addenda but CESSAR does not indicate which ASME Code Editions and Addenda are applicable for System 80 nuclear plants.

The ASME Code in the Summer 1972 Addenda instituted requirements for calibration of fracture toughness tests instruments and equipment. We have reviewed these requirements and consider that System 80 nuclear plants which meet this code addenda, have met the intent of Section III.B.5 of Appendix G, 10 CFR Part 50. Since material in CESSAR plants must be fracture toughness tested to at least the Summer 1972 Addenda of the ASME Code, they will meet the intent of the certification requirements of Section III.B.5 of Appendix G, 10 CFR Part 50 and an exemption to the specific certification requirements of Section III.B.5 may be granted.

Section III.C.2 states, in part, that the excess material for test specimens representing the reactor vessel beltline welds be prepared from actual production plates. CE has indicated that the weld test specimens may not be prepared from actual production reactor vessel plates, but will be prepared from plates of the same P number, same filler material and same production welding conditions as those used in joining the corresponding shell material. CESSAR has not indicated the welding conditions which will be the same for the test samples and the production welds. In order for us to grant an exemption to the requirements of Section III.C.2 the weld qualification conditions which must be the same for the test sample and production weld are those identified in NB 4330 of the 1971 Edition of the ASME Code. The weld qualification conditions for materials fabricated to the Summer 1972 Addenda are the same as those fabricated to the 1971 Edition of the ASME Code.

An exemption to not utilize actual production plates for preparation of reactor vessel beltline weld test samples is justified because weld properties are dependent on filler material and welding conditions.

Although CESSAR has not defined the weld conditions for simulation of production welds, CESSAR plants must meet the weld qualification conditions of NB 4430 of the 1971 Edition of the ASME Code, therefore, an exemption to the test sample requirements of Section III.C.2 may be granted.

CE has indicated in Section 5.2.3.3.1 of the CESSAR FSAR that the requested exemptions from Appendix G are applicable to the CESSAR scope. We will review the requested exemptions from Appendix G associated with the BOP scope, if any, in our review of each reference plant application. Those exemptions found acceptable for both CESSAR and the BOP scopes will be granted in the licenses for CESSAR reference plants.

Compliance to Appendix H, 10 CFR Part 50

The toughness properties of the reactor vessel beltline materials will be monitored throughout their service life by a materials surveillance program that meets the requirements of Appendix H, 10 CFR Part 50. CESSAR indicates that all System 80 reactor vessel surveillance will satisfy the requirements of Appendix H, 10 CFR Part 50.

Conclusions of Compliance to Appendices G and H, 10 CFR Part 50

We have reviewed CESSAR and determined that System 80 nuclear plants will comply with Appendices G and H except for Sections III.B.5 and III.C.2 of Appendix G. CESSAR plants meet the standards for an exemption to these requirements because the materials must be fracture toughness tested to at least the Summer 1972 Addenda of the ASME Code. Material fracture toughness tested to at least the Summer 1972 Addenda of the ASME Code will meet the intent of Sections III.B.5 and III.C.2 of Appendix G, 10 CFR Part 50.

Since these exemptions depend upon the reference plant meeting at least the Summer 1972 Addenda of the ASME Code, the reference plant must be reviewed to determine that the reference plant fracture toughness tests have been performed in acordance with the applicable ASME Code edition and addenda. This review will be reflected in the reference plant Safety Evaluation Report.

Although CESSAR indicates all System 80 nuclear plants will conform to the requirements of Appendix H, 10 CFR Part 50, each reactor vessel material surveillance program is designed based on the actual reactor vessel beltline fracture toughness properties. Since these properties vary in each plant, the actual reactor vessel beltline fracture toughness properties and material surveillance data must be reviewed for each reference plant to confirm that the applicants materials surveillance program complies with the requirements of Appendix H, 10 CFR Part 50. This review will be reflected in the reference plant Safety Evaluation Report.

Based on the foregoing, pursuant to 10 CFR Part 50, Section 50.12, the exemptions from the specific requirements of Appendix G of 10 CFR Part 50, as discussed above, are authorized by law and can be granted without endangering life or property or the common defense and security and are otherwise in the public interest. We conclude that the public is served by not imposing certain provisions of Appendix G of 10 CFR Part 50 that have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality of safety.

Furthermore, we have determined that the granting of these exemptions does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. We have concluded that these exemptions would be insignificant from the standpoint of environmental impact statement and, pursuant to 10 CFR 51.5(d)(4), that an environmental impact appraisal need not be granted in connection with this action.

Appendix G, "Protection Against Nonductile Failure," Section III of the ASME Code, will be used, together with the fracture toughess test results required by Appendices G and H, 10 CFR Part 50, to calculate the reactor coolant pressure boundary pressure-temperature limits, as described in Section 5.3.2 of this report. The fracture toughness tests required by the ASME Code and Appendix G of 10 CFR Part 50 will provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. The use of Appendix G, Section III of the ASME Code, as a guide in establishing safe operating procedures, and the use of the results of the fracture toughness tests performed in accordance with the ASME Code and NRC regulations, will provide adequate safety margins during operating, testing, maintenance, and anticipated transient conditions. Compliance with these Code provisions and NRC regulations constitute an acceptable basis for satisfying the fracture toughness requirements of GDC 31.

The materials surveillance program, required by Appendix H, 10 CFR Part 50, will provide information on material properties and the effects of irradiation on materials properties so that changes in fracture toughness of material in the reactor vessel beltline caused by exposure to neutron radiation can be properly assessed and adequate safety margins against the possibility of vessel failure can be provided. Compliance with Appendix H, 10 CFR Part 50, together with the standard ASTM E-185-73, by the applicants referencing CESSAR, will assure that the surveillance program constitutes an acceptable basis for monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and satisfies the materials surveillance requirements of GDC 31 and 32.

5.3.2 Pressure-Temperature Limits

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," 10 CFR Part 50, describe the conditions that require pressure-temperature limits for the reactor coolant pressure boundary and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins for the reactor coolant pressure boundary at least as great as the safety margins recommended in the ASME Code, Section III, Appendix G, "Proctection Against Nonductile Failure." Appendix G, 10 CFR Part 50, requires additional safety margins whenever the reactor core is critical, except for low-level physics tests.

The following pressure-temperature limits imposed on the reactor coolant pressure boundary during operation and test are reviewed to ensure that they provide adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components as required by GDC 31:

(1) Preservice hydrostatic test,

(2) Inservice leak and hydrostatic tests,

(3) Heatup and cooldown operations, and

(4) Core operation.

CE has not provided actual pressure-temperature limits for any CESSAR plants. However, the CESSAR FSAR Sections 5.3.2.1 and 5.3.2.2 have outlined a method for preparing pressure-temperature limit curves which are in conformance with Appendix G, 10 CFR Part 50. Upon receipt of the plant specific pressure-temperature limit curves, we will confirm in the reference plant SER that the plant-specific curves comply with the requirements of Appendix G, 10 CFR Part 50.

The pressure-temperature limits to be imposed on the reactor coolant system for all operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure must be in conformance with established criteria, codes, and standards acceptable to the staff. The use of operating limits based on these criteria, as defined by applicable regulations, codes, and standards will provide reasonable assurance that nonductile or rapidly propagating failure will not occur and will constitute an acceptable basis for satisfying the applicable requirements of GDC 31.

5.3.3 Reactor Vessel Integrity

We have reviewed CESSAR with respect to the reactor vessel integrity. Although most areas are reviewed separately, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted.

We have reviewed the information in each area to ensure that it is complete and that no inconsistencies exist that would reduce the certainty of vessel integrity. The areas reviewed are:

(1) Design (SER 5.1)

(2) Materials of construction (SER 5.3.1)

(3) Fabrication methods (SER 5.3.1)

(4) Operating conditions (SER 5.3.2)

We have reviewed CESSAR to the above factors and conclude that CESSAR reactor vessels will comply with Appendices G and H, 10 CFR Part 50, except for Sections III.B.5 and III.C.2 of Appendix G as previously discussed in Section 5.3.1.

Section III.B.5 of Appendix G requires that the fracture toughness results include a certification that the tests were preformed in accordance with the requirements of Appendix G, 10 CFR Part 50. Since CESSAR plants must be fracture toughness tested to at least the Summer 1972 Addenda of the ASME Code, they will meet the intent of the certification requirements of Section III.B.5 and an exemption to Section III.B.5 may be granted.

Section III.C.2 of Appendix G, in part, requires that excess material for weld test specimens representing the reactor vessel beltline welds be prepared from actual production plates. CESSAR plants are required to meet the weld qualification conditions of NB 4330 of the 1971 Edition of the ASME Code which specifies the weld conditions for preparation the test samples that assures the weld test sample and production weld are metallurgically equivalent. Since the CESSAR reactor vessel beltline test samples must comply with NB4330 of the 1971 Edition of the ASME Code, the test samples may be considered representative of production welds and an exemption to Section III.C.2 may be granted.

We have reviewed all factors contributing to the structural integrity of CESSAR reactor vessels and conclude there are no special considerations that make it necessary to consider potential reactor vessel failure.

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps

The reactor coolant pumps provide sufficient forced circulation flow through the reactor coolant system to assure adequate heat removal from the reactor core during power operation. A low limit on reactor coolant pump flow rate (i.e., design flow) is established to assure that specified acceptable fuel design limits are not exceeded.

The reactor coolant pump and motor assembly in conjunction with the flywheel, provide sufficient coastdown flow following loss of power to assure adequate core cooling.

5.4.1.1 Pump Flywheel Integrity

GDC 4, "Environmental and Missile Design Bases," of Appendix A, 10 CFR Part 50, requires, in part, that nuclear power plant structures, systems, and components important to safety be protected against the effects of missiles that might result from equipment failures. Because reactor coolant pump flywheels have large masses and rotate at speeds of approximately 1200 revolutions per minute during normal operation, a loss of flywheel integrity could result in high energy missiles and excessive vibrations of the reactor coolant pump assembly. The safety consequences could be significant because possible damage to the reactor coolant system, the containment, or the enginered safety features.

Adequate margins of safety and protection against the potential for damage from flywheel missiles can be achieved by the use of suitable material, adequate design, and inspection. The CESSAR flywheel materials will be produced by a process that will minimize flaws and improve fracture toughness, and will be cut, machined, finished, and inspected in accordance with Section III of the ASME Code and Regulatory Guide 1.14.

The CESSAR reactor coolant pumps will be designed for 125% of the normal sychronous speed of the motor (approximately 1500 rpm). However, the minimum speed for failure is estimated to be much higher than 125% of operating speed for flywheels of the CESSAR design. The CESSAR FSAR has stated that the minimum fracture toughness of the flywheel material at normal operating temperature will be equivalent to a dynamic stress intensity factor (K_{Id}) equal to or greater than 100 ksi/in.

CESSAR does not include an inservice examination program and does not report the actual material and fracture toughness data for each CESSAR flywheel. This information is plant specific and will be evaluated in the reference plant Safety Evaluation Report. Compliance with Regulatory Guide 1.14 will provide a basis acceptable to the staff for satisfyting the requirements of GDC 4.

5.4.2 Steam Generators

The two steam generators are designed to transfer 3817 MWt from the RCS to the secondary system, producing approximately 17.18 x 10^6 lb/hr of 1067 psia saturated steam, when provided with 450 degree Fahrenheit feedwater. Moisture separators and steam driers in the shell side of the steam generator limit the moisture content of the steam to 0.25 wt% during normal operation at full power.

Moisture-separating equipment in the shell side of the steam generators limit moisture content of the exit steam. Manways and handholes are provided for access to the steam generator internals. Reactor coolant enters at the bottom of each steam generator through the single inlet nozzle, flows through the U-tubes, and leaves through the two outlet nozzles. A vertical divider plate separates the inlet and outlet plenums in the lower head.

The steam generator with integral economizer is in most respects similar to earlier U-tube recirculating steam generators. The basic difference is that instead of introducing feedwater only through a sparger ring to mix with the recirculating water flow in the downcomer channel, feedwater is also introduced into a separate, but integral section of the steam generator. A semi-cylindrical section of the tube bundle, at the cold leg or exit and of the U-tubes, is separated from the remainder of the tube bundle by vertical divider plates. Feedwater is introduced directly into this section and pre-heated before discharge into the evaporator section.

The lower portion of the evaporator section and the downcomer channel occupy only one-half of the steam generator cross-section. The effect of this non-symmetry is considered in calculation of recirculation ratio, internal flow considerations, and in design of tube support structures.

The steam-water mixture leaving the vertical U-tube heat transfer surface enters the separators which impart a centrifugal motion to the mixture and separate the water particles from the steam. The water exits from the perforated separator housing and recirculates through the downcomer channel to repeat the cycle. Final drying of the steam is accomplished by passage of the steam through corrugated plate dryers.

5.4.2.1 Steam Generator Materials

The staff concludes that the materials specified for the steam generator are acceptable and meet the requirements of GDC 1, 14, 15, and 31 of Appendix A, and Appendix B of 10 CFR Part 50. This conclusion is based on the following considerations.

CE will meet the requirements of GDC 1 with respect to codes and standards by assuring that the materials for use in Class 1 and Class 2 components will be fabricated and inspected in conformance with codes, standards, and specifications acceptable to the staff. Welding qualification, fabrication, and inspection during manufacture and assembly of the steam generator will be done in conformance with the requirements of Sections III and IX of the ASME Code.

The requirements of GDC 14 and 15 have been met to assure that the reactor coolant pressure boundary and associated auxiliary systems will be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture during normal operation and anticipated operational occurrences.

The primary side of the steam generator is designed and fabricated to comply with Class 1 criteria of the ASME Code as required by the staff. The secondary side of the pressure boundary of the steam generator will be designed, manufactured, and tested to Class 1 criteria although the required classification is ASME Code Class 2.

The crevice between the tube sheet and the inserted tube will be minimal because the tube will be expanded to the full depth of insertion of the tube in the tube sheet. The tube expansion and subsequent positive contact pressure between the tube and the tube sheet will preclude a buildup of impurities from forming in the crevice region and reduce the probability of crevice boiling.

The requirement of GDC 31 will be met with respect to the fracture toughness of the ferritic materials since the pressure boundary materials of ASME Class 1 components of the steam generator will comply with the fracture toughness requirements and tests of Subarticle NB-2300 of Section III of the ASME Code. The materials of the ASME Class 2 components of the steam generator will comply with the fracture toughness requirements of Subarticle NC-2300 of Section III of the ASME Code.

The requirements of Appendix B of 10 CFR Part 50 will be met since the onsite cleaning and cleanliness control during fabrication conform to the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

Reasonable assurance of the satisfactory performance of steam generator tubing and other steam generator materials is provided by (a) the design provisions and manufacturing requirements of the ASME Code, (b) provision of requirements for secondary water monitoring and control, and (c) the limiting of condenser inleakage. The controls described above combined with conformance with applicable codes, standards, staff positions, and Regulatory Guides constitute an acceptable basis for meeting in part the requirements of GDC 1, 14, 15, and 31 of Appendix A, and Appendix B of 10 CFR Part 50.

5.4.2.2 Steam Generator Inservice Inspection

The CESSAR steam generator design has incorporated ready access and other provisions for routine inservice inspection. The inspection program, however, is the responsibility of the applicant's referencing CESSAR. We will review the steam generator inservice inspection program in each reference plant application.

5.4.3 Shutdown Cooling (Residual Heat Removal) System

The shutdown cooling system (SDCS) is within the CESSAR design scope and is used in conjunction with the main steam and main or auxiliary feedwater systems to reduce reactor coolant system (RCS) temperatures from normal operating temperatures to the refueling temperature.

Initially, heat is rejected from the steam generators to the condenser or atmosphere. When the RCS temperature and pressure have been reduced to approximately 350°F and 400 psia, the SDCS is put into operation to reduce the reactor coolant temperature to the refueling temperature and to maintain this temperature during refueling.

When the SDCS is in operation, the system takes its suction from each RCS hot leg via a system of parallel lines and valves forming redundant trains. From the discharge of the two pumps, a portion of the coolant is diverted to the shutdown cooling heat exchangers which are cooled by component cooling water. The diverted flow is then mixed with the main SDCS flow streams and discharged into the four reactor cold legs. No single active failure to the SDCS system can cause the total loss of shutdown cooling or restrict the cooling ability such that the RCS cannot be brought to or maintained at refueling temperature.

Besides cooldown and cold shutdown, the SDCS operates in several other modes. These are:

(1) Startup - connected to chemical and volume control system (CVCS), acting as an alternate letdown path to control reactor coolant system pressure.

(2) Refueling - used for refilling the refueling canal.

(3) Emergency core cooling - the low pressure safety injection (LPSI) pumps which drive the SDCS are aligned during power operation and hot shutdown for low pressure coolant injection into the RCS as an integral part of the emergency core cooling system (ECCS).

(4) Containment spray - during normal operation the containment spray pumps are aligned to discharge through the shutdown cooling heat exchangers. This is the required alignment for emergency operation following a loss-of-coolant accident (LOCA). During shutdown cooling, the heat exchangers are isolated from the containment spray system.

SDCS leak detection is discussed in Section 5.2.5. If onsite electric power is available and offsite electric power is unavailable, the SDCS is capable of cooling the RCS given a single active failure. Each of the two SDCS trains may be isolated independently from the other while allowing the nonisolated 100% capacity train to perform its safety function, which is in compliance with GDC 34.

The SDCS is designed to comply with Regulatory Guide 1.29, "Seismic Design Classification," Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid Systems Components," as discussed in Section 3.2 and 3.9 of this report.

While the SDCS is a seismic Category I system, the structure housing it is not within the scope of CESSAR. Each applicant referencing the CESSAR design must demonstrate that the SDCS is housed in a structure that is designed to withstand tornadoes, floods, and seismic phenomena in accordance with GDC 2.

All applicants referencing this design, who intend to place multiple units at the site, must demonstrate that no components of the SDCS are shared between units, in order to show compliance with GDC 5.

The SDCS is designed to provide adequate isolation between the SDCS and the safety injection tanks on the RCS when the RCS is above the design pressure of the SDCS (435 psig) as follows:

(1) There are two parallel paths of SDCS suction lines which are equipped with three remotely controlled valves in each line. Two isolation valves per line are located inside containment. Interface criteria in CESSAR require that the four isolation valves inside containment be powered by four independent power supplies in such a way that a fault in one power supply or valve will neither line up the RCS to either of the two SDCS trains inadvertently nor prevent the initiation of shutdown cooling with at least one train. Each applicant referencing CESSAR will be required to demonstrate compliance with this interface. In addition to the isolation valves, each suction line contains a motor-operated valve in the low pressure piping outside containment. Each applicant referencing CESSAR will be required to show that these valves are powered in such a way that a single failure of power supply or valve will not preclude initiation of the SDCS when considered along with the isolation valves. The four suction valves inside containment each have an independent interlock, utilizing pressurizer pressure, which prevent opening the valve when RCS pressure exceeds 400 psia. The interlocks also provide automatic closure of the valves when pressurizer pressure exceeds 500 psia.

(2) Safety injection tanks (SIT) will be isolated or depressurized prior to placing the SDCS into operation.

(3) There are two check valves and a closed motor-operated isolation valve on each line from the SDCS discharge to the cold legs to protect the system from RCS pressure. CE has provided design features to permit leak testing of each check valve to verify pressure isolation capability. These tests are discussed in Section 3.9.6 of this report.

Overpressure protection of the SDCS is provided by relief valves. Relief valves SI 169 and SI 469 are in the SDCS suction line between the isolation valves to protect isolated pipe lengths against transient thermal effects. Each valve has a 15 gallon per minute flow capacity and a setpoint of 2485 psig. Further protection is provided by relief valves SI 179 and SI 189 and located outside the containment isolation valves. These relief valves protect the SDCS from inadvertent RCS pressurization during SDCS operation. The valves are sized to protect the components and piping from overpressure due to inadvertent starting of the charging pumps, HPSI pumps, and pressurizer heaters. These valves have a set pressure of 435 psig and a capacity of 4000 gpm each. There are also relief valves at the discharge of each SDCS heat exchange to protect the system from pressure increases developed by temperature changes in component cooling water. The setpoint for these relief valves is 650 psig with a capacity of 120 gallons per minute each. Other isolated sections of piping also have small thermal relief valves, to protect from pressure increases due to the heating of trapped water. CESSAR requires that preoperational tests be conducted on each referenced plant to verify proper operation of the SDCS. The preoperational tests include testing of the automatic flow control, verification of adequate shutdown cooling flow, and verification of the operability of all associated valves. In addition, a preoperational hot functional performance test is made on the installed shutdown cooling heat exchangers. Flow tests complying with Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants," will be required of all applicants referencing CESSAR, during preoperational testing to verify the design performance of the system and its individual components. In addition, preoperational hydrostatic tests will be performed per Section III of the ASME Boiler and Pressure Code, while in service hydrostatic testing will be performed per Section XI of the ASME Code. Details of the preoperational testing program with the CESSAR scope can be found in Section 14 of this report.

During the course of our review, we requested that CE demonstrate how the requirements of Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," have been met by the CESSAR design. Specifically, CE was asked to demonstrate that the plant could be brought to SDCS initiation in less than 36 hours using only seismic Category I equipment, assuming the most limiting single failure, and with only onsite or only offsite power available. We requested that CE demonstrate that the seismic Category I auxiliary feedwater system required by CESSAR has sufficient inventory to maintain the plant at hot shutdown conditions for 4 hours, and then cooldown to the point where core decay heat could be rejected by the shutdown cooling system (SDCS). In addition, supporting analysis was requested which would:

(1) Confirm that adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions. The analysis must include an estimate of the times required to achieve such mixing, and

(2) Confirm that the cooldown under natural circulation conditions can be achieved within the limits specified in the emergency operating procedures.

The CE response identified the systems which would be used to meet these requirements. Cooldown to cold shutdown conditions employs the auxiliary feedwater system, the main steam system, and the shutdown cooling system. The initial plant cooldown is accomplished by heat rejection to the atmosphere by the steam generator atmospheric dump valves. Four safety-grade atmospheric dump valves, two per steam generator, are stated as an interface requirement. The atmospheric dump valves, valve operators, and power supplies must all be built in accordance with seismic Category I Quality Class II requirements. The valves must also be supplied with handwheels to allow them to be operated manually and, if air-operated, be supplied with a safety-grade air source. Should a single failure of an emergency power train occur making two atmospheric dump valves inoperable from the control room, the other two valves, one per steam generator, are sufficient for plant cooldown.

During loss of offsite power the reactor coolant system is depressurized using auxiliary spray. The auxiliary spray valves and associated charging pumps are safety-grade and have vital power supplied by emergency onsite power. Redundant auxiliary spray valves are provided.

Boration is accomplished using the chemical and volume control system. This system incorporates three charging pumps with redundant charging pump suction and delivery paths. This system satisfies the single failure criterion and can function without offsite power. However, to supply RWT water for makeup and boration to the changing pumps suction valves CH 141 and CH 501 must be operated manually if offsite power is not available.

These manual actions outside of the control room do not meet the requirements of Branch Technical Position 5-1, and are unacceptable. In a letter dated October 29, 1981, CE committed to provide motor operators which are powered from onsite emergency sources for these valves. We will report on the resolution of this issue in a revision to this report.

When the plant reaches the appropriate temperature and pressure, the shutdown cooling system is aligned and the cooldown proceeds by rejecting heat to the shutdown cooling system heat exhcangers. Prior to this, the safety injection tanks (SITs) must be secured or vented (from the control room) to permit depressurization of the primary system below the 600 psig setpoint of these tanks. CESSAR will include an interface requirement that the SI tank vent valves can be operated from the control room.

With the exception noted above, all of the actions necessary for shutdown can be performed from the control room. Sufficient controls and instrumentation are required by CESSAR to initiate and maintain the plant in shutdown cooling. Therefore, the SDCS meets the staff position, pending resolution of the identified deficiency.

Assuming the loss of offsite power, the most limiting single failure associated with cooldown is the failure of a dc bus and associated diesel generator. This failure disables the auxiliary spray valve and one train of components associated with the chemical and volume control system, the auxiliary feedwater system, the component cooling water system, and the shutdown cooling system. Redundant systems are available for removal of decay heat, depressurization, boration, makeup, and SDCS operation, though limited manual actions may be necessary outside of the control room to reposition valves which have lost power due to the diesel failure. With this limiting failure, CESSAR has demonstrated that the plant can be placed into cold shutdown, within 36 hours of a reactor trip, which meets staff requirements.

The staff requires that CESSAR demonstrate its ability to cool down using natural circulation in the primary system, including the adequacy of boron mixing during this mode. Natural circulation tests will be conducted during power ascension tests on the lead CESSAR plant. CE has referenced the boron mixing tests to be conducted during startup at San Onofre Units 2 and 3 as being applicable to CESSAR. The staff finds this acceptable, pending a favorable evaluation of the San Onofre test results. The staff will, however, require that CE submit a report following the San Onofre tests, documenting the acceptability and applicability of the San Onofre tests for CESSAR. This will be required prior to startup testing of the lead CESSAR plant. If acceptable tests have not been conducted at San Onofre, the lead plant must conduct both boron mixing and natural circulation tests during its power escalation program.

Branch Technical Position RSB 5-1 requires that a seismic Category I auxiliary feedwater supply be provided with sufficient inventory to permit operation at hot shutdown conditions for at least 4 hours, followed by a cooldown to the conditions permitting operation of the shutdown cooling system. The inventory needed for the cooldown shall be based on the longest cooldown time needed with either only onsite or only offsite power available with an assumed single failure. CE has an interface requirement that at least 300,000 gallons of condensate storage be available in order to accommodate an initial hold at hot standby for 4 hours following by a cooldown to SDCS initiation. The analysis presented in CESSAR shows this is sufficient for a normal natural circulation cooldown. However, recent natural circulation cooldown events at operating reactors have indicated the possibility of voiding in the upper head of the reactor vessel when the Technical Specification cooldown rate of 75°F per hour is maintained. CE has stated that emergency operating procedures will be modified so that plant cooldown under natural circulation conditions would not result in upper head voiding. Since it is expected that a slower cooldown or depressurization rate will be required to prevent voiding, the staff will require that CE confirm that the CESSAR design has sufficient emergency feedwater which is specified as an interface requirement, is available, taking into consideration this newly revised cooldown rate. The demonstration of sufficient feedwater must consider holding the plant at hot shutdown for 4 hours and then cooling down to the SDCS initiation conditions at a rate which will not induce voiding. This confirmatory information must be provided prior to the FDA. With the above requirement, the staff finds that the condensate system interfaces meet the position specified in RSB 5-1.

The staff will require each applicant referencing CESSAR to provide a means to protect the LPSI pumps from a loss of pump suction or discharge path when in shutdown cooling. These alarms provide assurance that problems with pump suction or discharge paths will be identified, and corrective action taken prior to pump damage. Each application will be required to provide individual control room low flow alarms for each LPI pump which will be powered from emergency sources.

Based on the discussion given above, we conclude that System 80 meets the requirements of Branch Technical Position 5-1 as appropriate for Class 2 plants, with the exceptions noted.

The applicant referencing CESSAR will be required to provide information related to pipe breaks or leaks in high or moderate energy lines outside containment associated with the RHR system when the plant is in a shutdown cooling mode. Each reference plant applicant must demonstrate that for all potential break locations and maximum credible leak rate, the plant operators would have at least 20 minutes after the time of first alarm to take action before flooding of essential redundant equipment needed for shutdown is predicted to occur. The reference plant applicant must also demonstrate that adequate core cooling can be maintained for at least 20 minutes following the first alarm to a leak in the shutdown cooling system.

We have evaluated the shutdown cooling system design and interface criteria presented in CESSAR. With the exceptions described above, we conclude that the CESSAR design meets the requirements of GDC 5 and 34, as well as conforming to the recommendations of Regulatory Guides 1.26, 1.29, and 1.68 as noted above. On these bases, we find the design of the SDCS acceptable, and the associated interface requirements both acceptable and complete, pending resolution of the issues which are summarized below.

(1) Motor operators must be added to valves CH 141 and CH 501, so that shutdown can be accomplished from the control room, assuming loss of offsite power.

(2) Demonstration that sufficient auxiliary feedwater is available for cooldown and depressurization without voiding in the primary system.

(3) Submittal of report documenting the acceptability and applicability of San Onofre Units 2 and 3 boron mixing tests.

5.4.4 Pressurizer Relief Tank (Reactor Drain Tank)

The nonsafety-related (Quality Group D, nonseismic Category I) reactor drain tank is within the CESSAR scope and is designed to receive and condense normal discharges from the primary system (pressurizer) safety valves without a release to the containment atmosphere. This is accomplished by discharging the pressurizer steam under water in the reactor drain tank. The water level is maintained manually. Level, temperature, and pressure indication and alarms are provided in the control room to alert the operator to the reactor drain tank conditions and the need for makeup. A nitrogen blanket is also provided in the tank to permit expansion of the entering steam and to control the tank atmosphere. The tank is designed to withstand full vacuum. Overpressure protection is provided by a rupture disc which opens to the containment. The tank is sized to receive and condense the steam from the maximum expected step load event. The rupture disc relief capacity is greater than the combined relief capacities of the primary safety valves.

The reactor drain tank is located inside the containment which provides protection against natural phenomena. CESSAR includes an interface requirement that referencing applicants locate the tank rupture disc beneath a concrete ceiling or foundation to shield safety-related components from possible disc fragments which may result when the disc ruptures. Failure of the tank does not affect the integrity of the RCPB nor does it affect the capability to safety shutdown the plant as it is located downstream of the pressurizer safety valves and missile protection is provided, and thus the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components for Nuclear Power Plants," and Position C.2 of Regulatory Guide 1.29, "Seismic Design Classification," are met.

Based on our review, we conclude that the reactor drain tank meets the requirements of General Design Criteria 2 and 4 with respect to the need for protection against natural phenomena and internal missile protection, as its failure does not affect safety system functions, and meets the guidance of Regulatory Guides 1.26 and 1.29 concerning its seismic and quality group classification and is, therefore, acceptable and the interface requirement, as described above, is acceptable and complete.

6 ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

6.1.1 Engineered Safety Features

The engineered safety features components are provided by both CESSAR and the reference plant BOP design. The materials from both sources are usually the same specification steels, or of very similar composition and properties such that, for all intent and purposes, they are identical. The materials in BOP are reviewed separately and will be discussed in the reference plant Safety Evaluation Report.

The materials selected for the CESSAR portions of the Engineered Safety Features satisfy Appendix I of Section III of the ASME Code, and Parts A, B, and C of Section II of the Code, and the staff position that the yield strength of cold-worked stainless steels shall be less than 90,000 psi.

The ASME Codes applicable to CESSAR allow waiving fracture toughness testing of ferritic materials in Class 2 systems in the Summer 1977 Addenda, which specifically requires fracture toughness testing for pressure retaining materials. Waiving of fracture toughness testing of ferritic materials in Class 3 systems was allowed until the ASME Code Summer 1978 Addenda, which specifically required fracture toughness testing of ferritic pressure retaining materials. For units built to ASME Code Editions/Addenda which require fracture toughness testing, the staff finds the requirements adequate. For units built to ASME Code Editions/Addenda which allowed waiving fracture toughness testing of ferritic materials, the staff recommends acceptance. The rationale for acceptance is based upon (1) results of impact testing by other applicants of the same specification steels, and (2) fracture toughness data presented in NUREG-0577. We conclude that the fracture toughness properties of the engineered safety features ferritic materials will satisfy the fracture toughness requirements of present ASME codes.

The controls on the pH and chemistry of the reactor containment sprays and the emergency core cooling water following a postulated loss-of-coolant or design basis accident, are adequate to reduce the probability of stress corrosion cracking of the austenitic stainless steel components and welds of the Engineered Safety Features systems in containment throughout the duration of a postulated accident to completion of cleanup. The controls on the use and fabrication of the austenitic stainless steel of the systems meet most of the requirements of Regulatory Guide 1.31, "Control of Ferrite Content of Stainless Steel Weld Metal," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Where the recommendations of these guides were not followed, the actions taken by CE have been approved by the staff for other plants. Fabrication and heat treatment practices that will be performed will provide added assurance that the probability of stress corrosion cracking will be reduced during the postulated accident time interval. The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on components of the Engineered

Safety Features are in accordance with Regulatory Guide 1.35, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel." The control of the pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, is in accordance with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and provides assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution resulting from corrosion of containment metal or cause serious deterioration of the materials in containment. The protective coating systems will be qualified by tests acceptable to the staff. This qualification provides reasonable assurance that the coating systems will not degrade the operation of the ESF by delaminating, flaking, or peeling. Conformance with the Codes and Regulatory Guides and with the staff positions mentioned above constitutes an acceptable basis for meeting the applicable portions of the requirements of GDC 16, 34, 35, 38, 41, and 44.

6.1.2 Organic Materials

CE has committed to use, for CESSAR's scope of supply, protective coating systems inside containment that meet the quality assurance guidelines of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," and the standards of ANSI N101.2, "Protective Coatings (Paints) for Light-Water Nuclear Reactor Containment Facilities," American National Standards Institute (1972), excluding components limited by size and/or exposed surface area whose consequences will be evaluated separately for hydrogen generation and solid debris. CE has not indicated the quantity of protective coatings that do not meet Regulatory Guide 1.54 or ANSI N101.2 associated with the excluded components; however, CE has stated that there are 9800 lbs of organic materials in the electrical cable insulation inside containment.

Based on the above evaluation, we conclude that the protective coating systems and their applications are acceptable and meet the requirements of Appendix B to 10 CFR Part 50. This conclusion is based on CE having met the quality assurance requirements of Appendix B to 10 CFR Part 50 since the coating systems and their applications meet the positions of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," and the quality assurance standards of ANSI N101.2, "Protective Coatings (Paints) for Light-Water Nuclear Reactor Containment Facilities." Also, the containment coating systems have been evaluated as to their suitability to withstand a postulated design basis accident (DBA) environment. The coating systems chosen by the applicant have been qualified under conditions which take into account the postulated DBA conditions.

We have reviewed the protective coating systems and organic materials used for CESSAR scope of supply inside containment. Our review is limited in scope since reference plants may use additional protective coatings and organic materials inside containment. We will review the protective coating systems and total quantities of organic materials inside containment in each reference plant application referencing CESSAR.

6.2 Containment Systems

Containment systems include the containment structure (and penetrations) and associated systems such as the containment heat removal systems, the containment isolation system, and the containment combustible gas control system. Except for containment isolation provisions associated with CESSAR fluid system lines penetrating containment, the CESSAR scope does not include containment systems. However, in order to fulfill containment functional requirements, the design of all containment systems must be properly integrated with the pertinent aspects of the CESSAR design. In accordance with this, CESSAR contains the mass and energy release analyses for a spectrum of postulated reactor coolant system and secondary system pipe ruptures within containment that are needed by the reference plant to determine the containment design pressure and temperature and the design pressures of the containment subcompartments. CESSAR also includes the calculation of the minimum containment pressure required in the performance capability studies on the CESSAR emergency core cooling system (ECCS). In addition, CESSAR includes in Appendix 6A a description of the CESSAR containment spray system.

6.2.1 Containment Functional Design

6.2.1.1 Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents

CESSAR provides mass and energy release data for a spectrum of postulated reactor coolant system (RCS) breaks for use by the reference plant in establishing the containment design basis pressure and temperature. The spectrum of RCS breaks analyzed in CESSAR is listed in CESSAR Table 6.2.1-1 and includes a hot leg break and cold leg breaks at the RCS pump discharge and pump suction. We have reviewed the spectrum of breaks and believe that it should be sufficient to establish the design basis accident for the containment. However, if the reference plant analysis indicates that the most severe break is outside CESSAR's spectrum of breaks, we will require that additional breaks be analyzed on an individual plant basis.

The mass and energy release data for the RCS breaks are calculated to maximize the pressure and temperature in the containment. For all RCS breaks loss of offsite power is assumed. Also, both maximum safety injection (no pump or power source failure) and minimum safety injection (failure of one diesel) cases are included for the RCS cold leg breaks

The mass and energy released to the containment from a RCS break (i.e., loss-of-coolant accident) is considered in terms of the following phases: blowdown, refill, reflood, post-reflood, and long term. The blowdown phase starts at the initiation of the postulated pipe break. During the blowdown phase, the primary coolant is being rapidly injected into containment. The blowdown phase ends when essentially all of the reactor coolant has been injected into containment. During refill, the ECCS water refills the reactor vessel to the bottom of the core. This phase is conservatively omitted in the CESSAR analysis. The reflood phase ends when the water level in the core region reaches a height that is sufficient to quench the core. During the post-reflood phase the dominant heat transfer process is the removal of steam generator secondary energy remaining at the end of reflood by the ECCS water leaving the core. The post-reflood phase ends when the steam generator secondary temperature has essentially reached equilibrium with the primary side temperature, and there is no longer a significant driving potential for secondary to primary system heat transfer. During the long-term phase, ECCS water boils at the containment pressure. Energy sources during this phase are decay heat and the cooling of all nuclear steam supply system (NSSS) metal.

There is an important distinction between hot leg breaks and cold leg breaks for loss-of-coolant accident (LOCA) post blowdown analyses. For a hot leg break, the majority of the ECCS supplied water leaving the core can vent directly to the containment through the break, without passing through a steam generator. Therefore, since there is no mechanism for rapidly releasing the steam generator energy to the containment for a hot leg break, only the blowdown period need be considered. Conversely, for cold leg breaks, the water must pass through a steam generator before reaching the containment so that post blowdown releases to the containment are more important and must be considered for cold leg breaks.

The mass and energy release during the blowdown phase of a LOCA is calculated in CESSAR by the CEFLASH-4 computer code. The staff has found this code to be acceptable for blowdown calculations. This finding was transmitted to CE by a letter dated June 13, 1975 and is based on the staff's evaluation of the CE Topical Report CENPD-26.

CE, in selecting input data for the blowdown mass and energy release calculations, utilized heat transfer parameters (e.g., conductivities and heat transfer coefficients) that are well accepted and tend to overpredict the heat transfer and thus energy release. Conservative maximum reactor coolant system water volumes were selected to ensure that the greatest amount of available mass and energy was in the system prior to the loss-of-coolant accident. Operation of plant equipment, such as turbine stop valve and main feedwater isolation valve closure times and main and emergency feedwater flow additions, were all conservatively estimated for both blowdown and reflood calculations. Initial reactor power was assumed at 102 percent of design, and decay heat was computed using normalized decay heat curves with an additional 20 percent conservatism factor for the first 1000 seconds and a 10 percent conservatism factor thereafter. We have reviewed these and other assumptions listed in CESSAR for calculating the blowdown mass and energy releases and additionally have checked these assumptions against the description of the blowdown model found acceptable by the staff in the CESSAR SER for the Preliminary Design Approval (PDA), dated December 1975. Based on this, we conclude that CE has calculated in CESSAR blowdown mass and energy release data that conservatively maximize the energy releases to the containment.

In CESSAR the mass and energy release to containment during both the reflood and the post-reflood phases of the accident are calculated using the FLOOD-MOD 2 computer code. This code was previously found acceptable by the staff in the CESSAR PDA SER, dated December 1975.

For the core reflood and post-reflood phases, CESSAR presents mass and energy release data for containment backpressures of 55 psia and 70 psia (typical for CESSAR plants) because the mass and energy release to the containment during reflooding is dependent on the containment pressure. Thus the reference plant can select the backpressure case that is consistent with the containment design. This information is provided to the reference plant as an interface requirement.

Additionally, during the reflood phase, CE made conservative sssumptions in the calculation of the mass and energy release including upper limit heat transfer coefficients to maximize heat transfer and carryout rate fractions (CRFs) in compliance with the Standard Review Plan Section 6.2.1.3 guidelines. Credit was assumed for the condensation of steam in the discharge legs by the cold ECCS water but only when the reactor vessel annulus was full and if the ECCS rate was great enough to thermodynamically condense all of the steam in the discharge legs. The post-reflood modeling assumptions were identical to the reflood modeling assumptions except in two respects. First, the carryout rate fraction was conservatively changed from 0.80 to 1.00 to increase system flow rates, and second, no credit for condensation in the discnarge legs was taken. Based on our review of the core reflood and post-reflood model presented in CESSAR and a comparison of these models against the ones found acceptable by NRC in the CESSAR PDA SER, dated December 1975, we conclude that mass and energy releases durng these two post blowdown phases have been calculated conservatively.

Following the post-reflood phase, the mass and energy source terms for the long-term phase include decay heat, heat transfer from primary metal to primary fluid, and energy released due to containment depressurization. Because long-term energy releases are containment design dependent, CESSAR states that the reference plant must generate the plant-specific long-term release data in its safety analysis report along with the basis for the plant-specific calculatior

Based on our review of CESSAR's calculation, we conclude that mass and energy releases to the containment resulting from loss-of-coolant accidents have been calculated in a conse: /ative manner and that the mass and energy release data presented in CESSAR Tables 6.2.1-2 through 6.2.1-10 are acceptable for use in the containment functional design analysis of the reference plants.

6.2.1.2 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment

CESSAR has calculated mass and energy releases to the containment for a spectrum of main steam line breaks (MSLBs). The spectrum of breaks listed in CESSAR Table 6.2.1-1 includes the largest slot and guillotine breaks at which a pure steam blowdown can occur at reactor power levels of 102, 75, 50, 25, and 0 percent. The selection of the break size at which pure steam blowdown occurs is conservative since it maximizes the resulting peak containment pressure and temperature for a given set of initial steam generator conditions. The breaks are conservatively assumed to be at the nozzle of one of the steam generators and the analysis takes credit for the flow restrictions in the nozzles of the steam generator. Main feedwater line breaks are not analyzed since such breaks result in a blowdown less limiting than the MSLB, because of their lower fluid enthalpy. Based on our review of the CESSAR spectrum of MSLBs we believe that these breaks should be sufficient to establish the design basis accident for the containment. However, if the reference plant analysis indicates that the most severe break case is outside the spectrum for which energy releases have been provided in CESSAR, we will require that additional break cases be analyzed on an individual plant basis.

The MSLB mass and energy release rates in CESSAR are calculated using the SGN-III computer code. This code was found acceptable by the staff in the SER for CESSAR, dated December 1975.

Following a postulated main steam line break, the contents of the affected steam generator will be released to the containment. Most of the contents of the unaffected steam generator will be isolated by the main steam isolation valves (MSIVs) and main feedwater isolation valves (MFIVs). In CESSAR a main steam isolation signal (MSIS) automatically closes the MSIVS and MFIVS upon receipt of either a steam generator low pressure signal or a containment high perssure signal. The specific closure times of the MSIVs and MFIVs are CESSAR interface requirements and are used in the MSLB analysis. In addition, the maximum volumes for fluid between the MSIVs and each steam generator (2000 cubic feet total for two steam lines), between the MSIVs and the turbine stop valves (14,000 cubic feet), and between the upstream MFIV and each steam generator (500 cubic feet) are considered in the MSLB analysis. These main steam and feedwater line volumes are included in CESSAR as interface requirements.

The CESSAR MSLB mass and energy release analysis assumed the availability of offsite power. This is conservative since it allows continuation of reactor coolant pump operation which maximizes the rate of heat transfer to the affected steam generator. Single failures that could affect the mass and energy release were also considered. There are two MFIVs in series in each feedwater line and if one fails the other provides isolation. The MSLB analysis has already included the flashing of the fluid in the lines from the upstream MFIV to the affected steam generator, hence the failure of a MFIV is already instrinsically included in the MSLB analysis. An MSIV failure was determined by CE not to be a credible event because the MSIVs are designed to minimize the probability of failure to close when required, i.e., the MSIVs have been designed to close based on a conservative calculation which maximizes the dynamic pressure loading on the valves for all possible flow rates and qualities and each valve has dual solenoid valves, received an actuation signal from both trains of MSIS actuation, fails closed upon loss of power, and is tested periodically. While this MSIV design may minimize the probability of failure to close when required, it does not preclude MSIV failure to close due to a postulated mechanical failure. However, since CESSAR requires the MSLB mass and energy release data be used with the assumption of a failure in the containment heat removal system and since, based on our experience, a single active failure in the containment heat removal system is more limiting than the single active failure of an MSIV to close, we find the assumption used by CE for the CESSAR mass and energy release data for MSLBs acceptable.

Based on the NRC staff review of the CESSAR calculations of MSLB mass and energy release, we find both the method of calculation and the mass and energy and energy release data presented in Tables 6.2.1-11 through 6.2.1-20 acceptable for reference plant use in determining the containment design pressure and temperature.

6.2.1.3 Subcompartment Analysis

CE has calculated the mass and energy release rates during the short-term period following a reactor coolant system or secondary system pipe break for use in reference plant analyses to establish the design bases for the containment subcompartments. CESSAR Table 6.2.1-24 summarizes the spectrum of postulated pipe ruptures for which mass and energy releases are calculated. The break areas of the main reactor coolant system pipes are limited based upon the implementation of a set of pipe whip restraints within the range of parameters tabulated in Table 4-1 of the Topical Report CENPD-168A (see also CESSAR Section 3.6.2). For all other piping (i.e., the pressurizer surge line, the pressurizer spray line, the pressurizer relief line, and the main steam line) full double-ended circumferential ruptures are assumed.

The containment subcompartment mass and energy release rates from all the postulated pipe breaks are calculated with the CEFLASH-4A computer code. For the reactor coolant system breaks, the CEFLASH-4A comppter code, the assumptions made to maximize the blowdown rate, and the nodalization scheme for the reactor coolant system have already been accepted by the staff in the SER for CESSAR MSLB modeling assumptions and the nodalization scheme of the main steam system, we also find as acceptable the CESSAR subcompartment mass and energy releases for the MSLB case.

The staff concludes that the method described in CESSAR for calculating mass and energy release rates for subcompartment analysis will produce conservative results. The mass and energy release presented in CESSAR Table 6.2.1-25 through 6.2.1-35, however, may not be appropriate for a particular subcompartment design. For example, a reference plant must satisfactorily demonstrate that the main reactor coolant system piping restraint design complies with the CESSAR interface requirements and topical report CENPD-168A as discussed in Section 3.6.2 of this report. CESSAR also states that the mass and energy releases calculated in CESSAR are generally not applicable for main steam line break locations other than at the steam generator nozzle. Therefore, if design features of the balance of plant invalidate the break sizes or locations analyzed in CESSAR for containment subcompartment analyses, we will require that the reference plant safety analysis report include new mass and energy releases, their calculational basis, and appropriate justification of the break sizes and/or locations analyzed.

2.1.4 Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System

Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the containment pressure used for evaluating cooling effectiveness during reactor core reflood shall not exceed a pressure calculated conservatively for this purpose. The calculation includes the effect of operation of all installed containment pressure reducing systems and processes. The resulting reflood rate in the core will then be reduced because lessened containment pressure reduces the resistance to steam flow in the reactor coolant loops and increases the boiloff rate from the core.

For the CESSAR ECCS minimum containment pressure calculation, the CEFLASH-4A computer program was used to determine the mass and energy released to containment during the blowdown phase of a postulated LOCA, and the COMPERC-II computer program was used to determine both the mass and energy released to the containment during the reflood phase and the minimum containment pressure response to be used in the evaluation of the effectiveness of the ECCS. The staff reviewed this calculational method and published a Status Report on October 16, 1974, which was amended on November 13, 1974. In the NRC's SER for the CESSAR PDA, dated December 1975, the staff concluded that this CE containment pressure calculation method was acceptable for ECCS evaluation.

Since the CESSAR ECCS performance analysis for the CESSAR standard plant is intended to be applicable to many reference plants, the plant dependent input parameters for the minimum containment pressure analysis were chosen so they would envelop the balance of plant designs and minimize the calculated containment pressure after a loss-of-coolant accident. These input parameters include initial containment atmosphere temperature (50°F, minimum), pressure (14.7 psia, minimum), humidity (100%, maximum), net free containment volume (3,707,000 cubic feet), and assumptions for active containment heat removal systems and passive heat sinks that provide maximum heat removal capability. We will require, in each application referencing CESSAR, a comparison of the balance of plant design parameters to the significant plant dependent parameters used in the CESSAR minimum containment pressure analyses to ensure that the reference plant parameters are enveloped by the CESSAR parameters.

In conclusion, we find that Combusion Engineering performed the minimum containment pressure analysis in accordance with the requirements of 10 CFR Part 50 Appendix K and the guidelines of Branch Technical Position CSB 6-1, "minimum Containment Pressure Model for PWR ECCS Performance Evaluation." The break analyzed was the 1.0 x DEG/PD (double-ended guillotine/pump discharge leg) which is the break that produces the highest peak clad temperature. The selection of plant dependent input parameters is conservative, but must be checked against the actual balance of plant design for each reference plant. Also, because the CESSAR minimum containment analysis assumed complete containment isolation, if the balance of plant design includes a purge/vent system that could be open during the operational modes of power operation, startup, hot standby, or hot shutdown, the effects of this on the minimum containment pressure analysis must be presented by the reference plant application.

6.2.2 Containment Spray System

The CESSAR Containment Spray (CSS) provides containment heat removal for the CESSAR NSSS in the unlikely event of an accident that releases mass and energy into the containment. The CSS is designed to provide sufficient heat removal capacity, in conjunction with other acceptably defined active and passive heat sinks described in the reference plant safety analysis report, to prevent exceeding containment design pressure and temperature and to reduce containment pressure to at least one half of the design pressure in twenty-four hours following a loss-of-coolant accident (LOCA), control element assembly ejection, or a main steam or feedwater line break inside containment. The safety evaluation of the CSS presented below is based on the CSS described in CESSAR FSAR, Appendix 6A.

The CSS consists of two independent 100%-capacity loops each containing a containment spray pump, a shutdown cooling heat exchanger, a spray header, and associated valves, piping, and instrumentation. Each of the two containment spray pumps is rated at 3650 gal/min, including the minimum bypass flow of 150 gal/min, at a head of 525 feet. A minimum flow line, running from each pump discharge to the refueling water tank (RWT), ensures that the pump is not deadheaded if it is inadvertently run against a closed system. Containment spray is automatically initiated by a containment spray actuation signal (CSAS), which occurs on high-high containment pressure. Upon receipt of a CSAS the containment spray pumps are started, the spray header isolation valves are opened, and borated water from the RWT is delivered to the containment spray headers. Rated flow from the nozzles is established within 50 seconds after receipt of a CSAS, assuming loss of the preferred plant electrical power source. When the water level in the RWT reaches a specified low setpoint, a recirculation actuation signal (RAS) will automatically isolate the minimum flow lines and align the containment spray pump suctions to the containment sumps by opening the containment sump isolation valves. The operator must then verify that the appropriate amount of water has been discharged to the containment, that the flow paths from the containment sumps to the suctions of the containment spray pumps are open, and that the minimum flow lines are isolated, before closing the RWT isolation valves to complete the transitioon from the injection mode to the recirculation mode. During both the injection and the recirculation modes, the containment spray pumps discharge through shutdown cooling heat exchangers that provide cooling of the spray water. Cooling water to the shutdown heat exchangers is an interface requirement, as described in Section 5.4.3 of this report, and will be described in the reference plant SAR. Sufficient CSS instrumentation and controls have been provided to allow the operator to adequately monitor CSS conditions and performance and to manually perform all required CSS safety functions.

The CSS satisfies the provisions of Regulatory Guide 1.26 "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants." and 1.29 "Seismic Design Classification." CESSAR has also provided a failure mode and effects analysis (FMEA) and other information demonstrating the ability of the CSS to function following postulated single active failures. In addition, CESSAR has provided results of containment spray nozzle tests verifying the performance of the containment spray nozzles.

CESSAR has specified interface requirements to be met by components or systems, which are not in the CESSAR scope, but upon which the CSS depends to meet its functional criteria (CESSAR Appendix 6A Section 7.0). Based on our review of these interface requirements, we conclude that they are acceptable and provide complete information to ensure that the CSS will meet the functional design criteria described above.

We conclude that the CSS satisfies the requirements of General Design Criteria (GDC) 38, 39, and 40 explicitly. However, the final acceptability of the CSS will be dependent on the staff's review of the reference plant SAR for compliance with the CESSAR CSS interface requirements, the requirements of GDC 50, and the provisions of Regulatory Guide 1.1, "Net Positive Suction Head For Emergency Core Cooling and Containment Spray Systems." Compliance with GDC 50 would be based on the staff review of the physical arrangement of the spray headers and nozzles in containment and on the reference plant containment functional analysis which would verify the adequacy of the overall containment heat removal systems including the CSS.

6.2.3 Secondary Containment Functional Design

The CESSAR design scope does not include containment building design. The secondary containment functional design will be addressed in the reference plant safety evaluation, if appropriate.

6.2.4 Containment Isolation System

The function of the containment isolation system (CIS) is to allow the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to minimize the release of fission products that may result from a postulated accident, such as a loss-of-coolant accident (LOCA) or a fuel handling accident inside containment. In general, for each penetration at least two barriers are required between the containment atmosphere or reactor coolant system (RCS) and the outside atmosphere, so that the failure of a single barrier does not prevent isolation. Containment penetrations for fluid systems within CESSAR's scope are listed in CESSAR Table 6.2.4-1 and include lines of the safety injection system (SIS), shutdown cooling system (SCS), and chemical and volume control system (CVCS). Only the containment isolation provisions for these penetrations are reviewed here. All other containment penetrations or modifications of the penetrations covered in CESSAR, will be reviewed as part of the safety evaluation of the reference plant SAR.

Automatic containment isolation of penetrations within CESSAR's scope is accomplished by a containment isolation actuation signal (CIAS) and/or a safety injection actuation signal (SIAS) both of which are initiated by either high containment pressure or low pressurizer pressure. These isolation actuation signals can also be initiated manually from the control room. Documentation in CESSAR has demonstrated that each fluid system line that must be isolated immediately following an accident is automatically isolated by either a CIAS or SIAS or is sealed closed. Only essential lines (i.e., engineered safety feature (ESF) or ESF-related lines) or lines where other safety considerations require them to remain open (i.e., the CVCS charging line and reactor coolant pump seal water injection line) are not automatically isolated. All these lines, however, can be remote-manually isoalted. We conclude that adequate diversity has been provided with regard to the different monitored parameters that actuate containment isolation and that all nonessential lines, except the CVCS charging line and reactor coolant pump seal water injection line which are discussed below, are sealed closed or automatically closed when containment isolation is required.

We have reviewed the closure times for the power-operated containment isolation valves. Valve closure will occur within 60 seconds for all power-operated containment isolation valves and withon 10 seconds or less for most power-operated containment isolation valves, except the valves in the shutdown cooling system (SCS) suction lines, which have a maximum closure time of 80 seconds. We find this acceptable since each SCS suction line has two motor-operated gate valves in series which are locked closed during normal operation so they do not have to close to isolate the containment following an accident. We conclude that the containment isolation valve closure times for the CESSAR CIS are acceptable.

We have reviewed the designation of fluid system lines as essential or nonessential for penetrations that are within CESSAR's scope. Those systems or portions of systems classified as essential are the high and low pressure safety injection systems and the two containment sump suction lines. Two other lines, the CVCS charging line and the reactor coolant pump seal water injection line, although designated nonessential, are opening during normal operation and do not automatically close upon an isolation actuation signal. Cessar justifies this by stating that it is desirable to leave these two lines open to provide charging capability and reactor coolant pump seal injection capability after an accident in which offsite power is available. Consersely, it would be undesirable to lose charging or pump seal injection capability during normal operation due to an inadvertent isolation actuation signal. In addition, four factors mitigate the release of fission products through these two penetrations. These factors are: (1) flow through these lines is into containment and the reactor coolant system, (2) the lines having check valves inside containment to prevent backflow, (3) the connecting portions of the CVCS outside containment are designed to Safety Class 2 and seismic Cateogry I standards with a design pressure in excess of the containment design pressure, and (4) the operator can remote-manually isolate the lines from the control room if necessary. We find that the charging and reactor coolant pump seal injection lines may be important during an accident and that the isolation barrier design of these two lines is acceptable. However, we will require that Class IE emergency power be provided to the containment isolation valve CH-255 and CH-524. In a letter dated October 29, 1981, CE committed to provide a vital power interface requirement for these valves. We will report on the resolution of this issue in a revision to this report.

Our review of CESSAR has also found that as required each automatically closed isolation valve is provided with a remote-manual switch in the control room, all power-operated valves have position indication in the control room, each air-, hydraulic-, or electric-solenoid-operated isolation valve assumes the position of greater safety in the event of power failure to the valve operator, and sealed closed isolation valves are sealed closed in accordance with SRP Section 6.2.4, II.3.f. In addition, all pertinent containment penetrations have been provided with diverse power supplies for redundant isolation valves in series, except for the two containment sump suction line penetrations and the two shutdown system suction line penetrations. For these penetrations, safety requirements dictate that the electrical independence of system lines take procedence over the diversity of isolation valve power supply in a given line. However, in both cases the system outside containment is a closed system. Therefore, we find the CESSAR provisions for power source diversity acceptable.

Our review has confirmed that the CESSAR containment isolation provisions either meet the explicit requirements of GDC 55, or are aceptable on another defined bases as described above. The CESSAR scope does not include any penetrations subject to GDC 57 provisions.

The explicit requirements of GDC 56 are met except for the two containment sump suction line penetrations. For each containment sump suction line penetration, both isolation valves are located outside containment since it would be impractical to locate a valve inside containment. However, as suggested in the SRP in Section 6.2.4, the valve nearest containment for each penetration and the piping between the penetration and the valve are required by CESSAR to be enclosed in a leak tight or controlled leakage housing. Additionally, these lines, which are designed as essential, are locked closed from the control room during normal operation and are connected to a closed system outside containment for additional isolation protection. We find the design of the containment sump suction line penetrations acceptable on this basis and therefore find CESSAR's CIS in compliance with GDC 56.

The CESSAR CIS satisfies the provisions of Regulatory Guides 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and 1.29, "Siesmic Design Classification."

CESSAR has specified interface requirements to be met by components or systems which are not in the CESSAR scope, but on which the CIS depends to meet its functional criteria. The interface requirements are listed in CESSAR Table 1.2-2. In addition to these, the reference plant applicant must confirm that the physical locations of the outside isolation valves are as close to containment as practical and that there are acceptable leak detection provisions for all lines with nonautomatic remote manual isolation valves. Based on our review of the CESSAR interface requirements, we conclude that CESSAR, together with the above two additional requirements, provides acceptable and complete information on required interfaces to ensure the CIS functional criteria and the explicit requirements of GDC 54 are met.

We have also reviewed CESSAR with respect to the requirements of NUREG-0737 (see Section 22) item II.E.4.2, "Containment Isolation Dependability." Requirements pertaining to diversity of actuation signal parameters, designation of essential and nonessential systems, and automatic isolation of nonessential systems have been discussed above and determined acceptable. Requirements for containment setpoint pressure and containment purge and vent valves are not withn the scope of CESSAR and will be reviewed as part of the reference plant evaluation. Concerning the requirement dealing with the design of control systems for automatic containment isolation valves, CESSAR has stated that resetting the isolation signals will not result in the automatic reopening of containment isolation valves, that reopening will require deliberate operator action, and that reopening must be performed on a valve-by-valve or line-by-line basis. Thus, we find that for items within the scope of CESSAR the requirements of NUREG-0737 Item II.E.4.2 have been met.

We conclude that the containment isolation system for the containment penetrations within CESSAR's scope (see CESSAR Table 6.2.4-1) meets the requirements of General Design Criteria 54, 55, and 57, satisfies the provisions or Regulatory Guide 1.141, and conforms to all staff positions and industry codes and standards, and is therefore acceptable. The only exception concerns confirmation of the Class IE emergency power interface to valves CH-524 and CH-255.

6.2.5 Combustible Gas Control System

The CESSSAR design scope does not include the containment combustible gas control system. The combustible gas control system will be addressed in the reference plant safety analysis report.

6.2.6 Containment Leakage Testing Program

The containment leakage testing program will be addressed in the reference plant safety analysis report.

6.3 Emergency Core Cooling System

The emergency core cooling system (ECCS) is designed to provide core cooling as well as additional shutdown capability for accidents that results in significant depressurization of the reactor coolant system. These accidents include failure of the reactor coolant system piping up to and including the double-ended break of the largest pipe, breaks in the main steam piping, a CEA ejection accident and steam generator tube rupture.

The design criteria associated with the ECCS design are: 10 CFR 50.46, GDC 2, 4, 5, 17, 20, 27, 35, 36, 37, 54, 56, and the recommendations of Regulatory Guides 1.1, 1.11, 1.29, 1.47, 1.68, 1.79.

6.3.1 System Design

The emergency core cooling system consists of active and passive injection systems. The passive system (safety injection tanks) is actuated when the reactor coolant system (RCS) pressure drops below a preset value. The active components of the ECCS are the high pressure safety injection (HPSI) system and the low pressure safety injection (LPSI) system that are actuated by the safety injection actuation signal (SIAS).

The four safety injection tanks contain borated water covered by nitrogen pressurized to at lesat 600 psig. When the RCS pressure falls below the tank pressure, borated water is forced form the tanks into the four cold legs.

The HPSI mode of operation, upon actuation of the SIAS, consists of the operation of two high head centrifugal pumps which provide high pressure injection of borated water from the refueilng water tank (RWT) into the RCS. The charging pumps also align for injection following an SIAS to inject concentrated boric acid to the RCS. However, the charging flow has not been taken credit in the LOCA analysis.

Low pressure injection consists of two LPSI pumps which take their suction from the RWT. The RWT has a minimum volume of 502,760 gallons of borated water. A comparison between the ECCS equipment for CESSAR system 80 plant and at Waterford 3 is presented in Table 6.3-1.

The ECCS recirculation mode is automatically initiated by a Recirculation Actuation Signal (RAS). The RAS is actuated by a low RWT level signal which opens the containment emergency sumps discharge valves and terminate the LPSI pump operation. During the recirculation mode of operation the HPSI pump operation. During the recirculation mode of operation the HPSI pumps are operated with water supply from the emergency sumps. CE has specified interface requirements for the emergency sump design of reference plants.

Table 6.3-1 Emergency Core Cooling System Equipment Comparison

	Waterford 3	CESSAR
Low Pressure Safety Injection Pumps	2	2
Design Flow (gallons per minute)	4,050	4,200
Design Head (feet)	342	335
High Pressure Safety Injection Pumps	3	2
Design Flow (gallons per minute)	380	815
Design Head (feet)	2,830	2,850
Safety Injection Tanks	4	4
Design Pressure (psig)	700	700
Water Volume, Normal (cubic feet)	1,742	1,858
Refueling Water Tank	1	1
Water Volume, Minimum (gallons total)	443,000	502,760

6.3.2 Evaluation

The staff has reviewed the system description and piping and instrumentation drawings to assure that abundant core cooling will be provided during the initial injection phase with and without offsite power and assuming a single failure. The cold leg safety injection tanks have normally open isolation valves in their discharge lines. These valves will have power removed from the motor operators to preclude undetected closure during normal operation and inadvertent closure during the ECCS injection phase. There are two independent active injection trains. Each train contains a HPSI pump, a LPSI pump and its associated suction and discharge valves. The pumps and valves in the redundant trains are connected to separate power supplies. At least one train of injection pumps would be actuated assuming a single faliure in the power supply systems. An FMEA was presented by CE covering mechanical equipment in the ECCS. This analysis concluded that no single active or passive failure could prevent the ECCS from fulfilling its short- and long-term functions.

Electrically powered components of the ECCS, required for safety-related operation, can operate from onsite or offsite power in compliance with GDC 17. Components include pumps, valves, and instrumentation. Power must be removed from certain components during specific modes of operation to ensure plant safety. The following valves are locked in position and have their power locked out under the stated conditions.

(1) Safety injection tank isolation valves SI-624, SI-624, SI-634, SI-644 will be locked open when RCS pressure exceeds 700 psig in order to preclude the loss of a safety injection tank from a closed isolation valve during LOCA.

(2) Safety injection tank vent valves SI-605, SI-606, SI-607, SI-SI-608, SI-613, SI-623, SI-633, SI-643 will be locked closed with power locked out during plant normal operation. Each SIT has two parallel vent valves with each with each valve powered from separate emergency power sources. Additionally, each vent valve has an individual hand switch control in the control room.

There are two motor-operated isolation valves in series in the recirculation line of each ECCS train. These isolation valves are designed to close during recirculation mode while the ECCS pump suctions are transferred from RWT to containment emergency sumps.

All valves in the injection paths not receiving an SIAS signal are maintained locked in the open position by administrative controls. Actuator-operator valves are provided with key-operated control switches to prevent unintentional misalignment of safety injection flow paths during power operation. CE has identified, in response to staff questions, a list of six manual valves that are locked open by administrative procedures and the valve positions are not indicated in the control room. However, if any of those manual valves are improperly aligned, the improper alignment could only prevent flow from one train of the injection system. CE states that these six manual valves will not be moved from their locked-open positions except for repairs or maintenance on the safety injection pumps. Maintenance or repair is not expected on this system more than once a year. We agree that administrative procedures are sufficient to assure proper valve alignment for these six valves and conclude that the design of the safety injection valving system is consistent with

Regulatory Guide 1.47 and, therefore, valve position indication for those six manual valves are not required.

The LOCA analysis is described in Section 15.3.8 of this report. The results of staff review of these analyses are also presented in Section 15.3.8 of this report.

In response to the staff's question regarding the capability of the HPSI pumps to operate for extended periods of time, CE stated that the HPSI pump design is similar to steam generator feedwater pumps manufactured by Ingersoll Rand. CE provided feedwater pump operating data which showed that those pumps could be operated without pump overhaul for more than five years. CE also stated that the pumps will be inspected, and a recommendation for parts that should be replaced periodically. The HPSI pumps are expected to have major maintenance performed at significantly longer time intervals than feedwater pumps because the actual HPSI pump operation will be minimal. The routine inservice inspections defined in the technical specifications will verify that the performance of the HPSI pumps is acceptable.

The ECCS is designed with satisfactory high/low pressure isolation protection. The LPSI system is protected from RCS operating pressures by a closed motor-operated isolation valve in series with two check valves between LPSI discharge and each of the four vessels cold legs. The motor-operated isolation valve opens automatically on a SIAS. The HPSI system is a high pressure system, but is isolated from the RCS by a closed motor-operated isolation valve (open on SIAS) in series with two check valves in the line to each of the four cold legs. Both the LPSI and HPSI systems have relief valves to provide pressure relief for water trapped between closed valves should there be a temperature rise.

The environmental qualification of equipment in the ECCS is discussed in Section 3.11 of this report. All motor-operated valves and all pumps in the ECCS required to operate following a LOCA are located outside containment, with the exception of the SDCS isolation valves.

The recirculation actuation signal (RAS) automatically transfers suction of the HPSI and containment spray pumps from the RWT to the emergency containment sump and shuts off the LPSI pumps. The RAS meets NRC's single failure requirements as discussed in Section 7.3 of this report and actuates on a low RWT level signal. The design of the recirculation system is based on that one HPSI pump will be manually turned off by control room operator after startup of both pumps.

CE has specified interface requirements for the available NPSH required for HPSI and LPSI pump operation during various modes. The values specified are based on the required NPSH from the test data supplied by the pump manufacturer. We have reviewed the above test data and find the interface requirements contain NPSH requirements with sufficient margin.

CE has proposed a method of providing simultaneous hot and cold leg inejction to begin two hours following a LOCA to preclude an unacceptable boron concentration buildup in the core which might cause boron precipitation and reduction in core cooling. This is more than adequate time to make the valve realignment necessary to switch to simultaneous hot and cold leg HPSI.

All ECCS lines, including instrument lines, have suitable containment isolation features that meet the requirements of GDC 56 and RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment," as discussed in Section 6.2.4. The ECCS has no shared components between units in compliance GDC 5.

The ECCS is designed to comply with Regulatory Guide 1.29, "Seismic Design Classification."

The instrumentation needed to monitor and control the ECCS equipment following a LOCA has been reviewed. This instrumentation provides sufficient information for the operator to maintain adequate core cooling following an assumed LOCA. Post-accident monitoring instrumentation includes pressurizer pressure and level, steam generator pressure and level, HPSI and LPSI header pressure, reactor coolant temperature, containment pressure, RWT water level, SIS flow, and containment sump level.

In response to staff concern related to the ECCS pump protection, CE has confirmed that low flow alarms are being provided to the LPSI pump discharges and the HPSI pump discharge headers. CE has specified requirements for an operating procedure to reset the HPSI header flow alarm setpoints when one HPSI pump is manually shutoff after initiated by SIAS. These flow alarms will have emergency power supplies. We find this acceptable.

CE specifies the ECCS balance of plant interface requirements in Section 6.3.1.3 of the CESSAR FSAR with regard to power supply, protection from natural phenomena, missiles, systems separation and independence, monitoring, inspection and testing, chemistry/sampling, materials, system/component arrangement (including containment sump), radiological waste, related services, and environment.

CESSAR does not identify the specific insulation materials used on the systems and components in the CESSAR scope. Consequently, the applicants referencing CESSAR must identify all of the insulation materials used inside containment and demonstrate conformance with the guidelines of Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems."

6.3.3 Testing

The applicant will demosntrate the operability of the ECCS by subjecting all components to preoperational and periodic testing, consistent with RG 1.68, "Preoperational and Initial Startup Test Programs for Water Cooled Power Reactors," and RG 1.79, "Preoperational Testing of Emergency Core Cooling System for Pressurized Water Reactors, and GDC 37. The details of the applicant proposed tests are evaluated in Section 14 of this report.

6.3.4 Conclusions on the Emergency Core Cooling Systems

The staff concludes that the ECCS proposed by CESSAR is acceptable because it meets GDC 2, 4, 5, 17, 20, 27, 30, 37, 54, 56, and Regulatory Guides 1.1, 1.11, 1.29, 1.47, 1.68, and 1.79.

The staff evaluation with regard to the acceptance criteria of 10 CFR 50.46 is addressed in Section 15.3 of this report.

We have reviewed the interface requirements for the ECCS systems in the context of the Palo Verde reference plant application and found them acceptable. However, we have not, as yet, completed our review with regard to the acceptability and completeness of the interface requirements for future reference plant applications. We will report on the resolution of this issue in a revision to this report.

6.4 Control Room Habitability

The CESSAR design scope does not include control room design. This matter will be addressed in the evaluation of reference plant applications.

6.5 Containment Spray as a Fission Product Cleanup System

The iodine removal system for the CESSAR is described in the "Iodine Removal System Licensing Report." A complete description of the containment spray system is contained in the CESSAR FSAR in an appendix to Chapter 6.

The iodine removal function of the containment spray system is achieved by adding trace levels of hydrazine, to enhance iodine removal, to the two redundant containment spray trains. Disodium or trisodium phosphate is recommended by the CESSAR interface requirements to be present in baskets in the containment sump for long-term pH control of the sump water.

Upon automatic initiation via the containment spray actuation signal, the system is designed to draw the boric acid solution from the refueling water tank (RWT). Positive displacement pumps will automatically start and add hydrazine from the hydrazine tank to the containment spray lines. The hydrazine concentration in the spray solution is maintained between 50 ppm and 63 ppm for at least 4 hours. When the RWT reaches low level, an automatic recirculation actuation signal switches suction from the RWT to the containment sump.

Following depletion of the hydrazine, the favorable partitioning of iodine is maintained by the recommended addition of disodium or trisodium phosphate to the boric acid solution in the containment sump. This buffer is stored in powder form in "baskets" in the containment sump, and dissolves when the water level in the sump reaches the baskets. The CESSAR interface requirements call for the post-LOCA sump solution pH to be at least 7.0. SRP Section 6.5.2 states, "A pH value exceeding 8.5 (for the sump solution) provides assurance that significant evolution of iodine does not occur."

Sump additive designs resulting in a lower post-LOCA sump pH than 8.5 but greater than 7.0 may be acceptable if the offsite radiological consequences due to the release of iodine (corresponding to the resulting liquid-vapor iodine partitioning) results in LPZ doses within the 10 CFR Part 100 guidelines. CESSAR will be acceptable, therefore, if the CE interface requirement for the long-term sump pH is raised to 8.5. However, if this interface requirement is not changed to 8.5, then the acceptability of the spray system will be established in our evaluation of the LOCA radiological consequences for each reference plant.

7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

The evaluation addresses the CESSAR instrumentation and control systems. Supplementary and auxiliary support instrumentation and control systems will be evaluated in the applications that reference CESSAR.

There are extensive interfaces between the instrumentation and control systems within the CESSAR and reference plant (BOP) scopes. Specific interface criteria for the instrumentation and control systems are enumerated in a set of CE documents designated "NSSS Interface Design Requirements for (plant name) for System 80 Standard Design, Criteria No. Systems 80-ICE-(document number)." Based on our audit of this document for the Palo Verde Nuclear Generating Station, we conclude that there is reasonable assurance that the procedure for defining interface requirements is acceptable and will result in complete identification requirements and conforms to the applicable regulatory requirements, as discussed in the remainder of this section, for each reference plant application. Consequently, we will require that reference plant applications identify exceptions to the interface criteria for safety-related systems contained in these documents.

The staff review and conclusions in this section are based on the material contained in the CESSAR FSAR, information presented at an Independent Design Review meeting, during which a transcript was kept and entered on the CESSAR docket, our audit of the CESSAR interface documents, and electrical drawings referenced in Chapter 1.7 or contained in quality assurance files in CE's offices.

7.1.1 Acceptance Criteria

The bases for evaluation of the CESSAR design, design criteria, and design bases are set forth in the Standard Review Plan (SRP), NUREG-0800, in Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems." These acceptance criteria include the applicable General Design Criteria (Appendix A to 10 CFR Part 50) and IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations" (10 CFR Part 50.55a(h)). Guidelines for implementation of the requirements of the acceptance criteria are provided in the IEEE Standards, Regulatory Guides, and Branch Technical Positions (BTPs) of the Instrumentation and Control Systems Branch (ICSB) identified in Section 7.1 of the SRP. Conformance to the acceptance criteria provides the bases for concluding that the instrumentation and control systems meet the requirements of 10 CFR Part 50.

7.1.2 General Findings

CE has identified the instrumentation and control systems important to safety. The acceptance criteria, consisting of the General Design Criteria (GDC) and IEEE Standard 279, are included in the Commission's regulations and are applicable to the systems as identified in the SRP. In addition, the applicant has identified the guidelines, consisting of the regulatory guides and the industry codes and standards which are applicable to the systems. The acceptance criteria and guidelines identified by the applicant are provided in Chapter 7.1 and Appendix A of CESSAR and Exhibit III of the Independent Design Review (IDR) transcript.

Based on the review of Section 7.1 of the applicant's FSAR, we conclude that the implementation of the identified acceptance criteria and guidelines satisfies the requirements of General Design Criterion (GDC) 1, "Quality Standards and Records," with respect to the design, fabrication, erection, and testing to quality standards commensurate with the importance of the safety functions to be performed. We find that the instrumentation and control systems important to safety, addressed in Section 7.1 of CESSAR, satisfy the requirements of GDC 1 and, therefore, are acceptable.

7.1.3 Technical Specification Requirements

Items to be included in the CESSAR technical specifications are discussed in the following sections of this report:

(1) Supplementary Protection System Operability (7.2.2)

(2) Reactor Trip Switchgear System Testing (7.2.4)

(3) Reactor Protection System and Engineered Safety Feature Actuation System Channel Bypass (7.2.7.2) (7.3.5)

(4) Emergency Feedwater Actuation Signal Channel Bypass (7.3.3)

In addition, reference plant technical specifications are to include:

(1) The range of CPC addressable constants,

(2) CPC cabinet temperature is to be monitored and functional testing performed should the CPC environmental temperature exceed the qualification temperature.

7.1.4 Reference Plant License Conditions

Items to be included as licensing conditions are discussed in the following sections of this report:

(1) Core Protection Calculator Software Change Methodology (7.2.1.3)

(2) Reactor Protection System and Engineered Safety Feature Actuation System Setpoint Methodology (7.2.6) (7.3.6)

In addition, reference plant licenses are to include the following conditions.

(1) CPC installation and operational issues originally raised during the ANO-2 CPC review are to be addressed by reference plant applicants and suitable plant procedures written.

(2) Procedures are to be written to test and protect application programs, run on the plant monitoring system, which are used to confirm conformance with Limiting Conditions for Operation defined in the plant Technical Specification.

(3) Plant protection system response time testing is to be performed using methods found acceptable by the staff.

(4) The reference plant shall provide alarms to ensure that the shutdown cooling system is aligned in the low temperature overpressure protection mode when the reactor coolant system is below the pressure-temperature operating limit (7.6.1).

7.1.5 Site Audit

A site review of each reference plant will be performed for the purpose of confirming that the physical arrangements and installation of electrical equipment are in accordance with the design criteria and descriptive information reviewed by the staff. The site review will be completed prior to issuance of the reference plant license and any problems found will be addressed in a supplement to the reference plant safety evaluation report.

7.2 Reactor Coolant System

The plant protective system (PPS) consists of a reactor protective system (RPS), a supplementary protective system (SPS), and the engineered safety features actuation system (ESFAS). The RPS and SPS are discussed below. ESFAS is discussed in Section 7.3. CESSAR standard scope of supply extends from process electronics (Foxboro or Westinghouse) through reactor trip switchgear to control rods, exclusive of field cabling.

On an instrument-by-instrument basis, transducers and associated equipment are also provided. Impulse lines are provided by the reference plant. Staff review of all equipment upsteam of the process electronics will be performed on a plant-specific basis, independent of supplier.

7.2.1 System Description

The functions of the RPS are (1) to initiate automatic protective action (reactor trip) to assure that fuel design limits and other safety limits are not exceeded during design basis incidents of moderate frequency and infrequent incidents and (2) to initiate automatic protective action (reactor trip) in conjunction with the ESFAS to limit the consequences of the design basis limiting faults.

The RPS monitors selected parameters and trips the reactor whenever established operational limits are reached. The trip parameters are:

(1) Variable overpower
(2) High logarithmic power level
(3) High local power density
(4) Low departure from nucleate boiling ratio
(5) High pressurizer pressure
(6) Low pressurizer pressure

(7) Low steam generator 1 water level
(8) Low steam generator 2 water level
(9) Low steam generator 1 pressure
(10) Low steam generator 2 pressure
(11) Variable low reactor coolant flow
(12) High containment pressure
(13) High steam generator 1 water level
(14) High steam generator 2 water level
(15) Manual trip

Four protection channels are provided for each of the trip parameters listed above. Trip parameters (3) and (4) are calculated trips provided by the digital core protection calculators. Trip parameters (1), (2) and (5) through (14) are derived from nuclear and process measurement signals, and trip parameter (15) is provided by switches in the control room and at the reactor trip switchgear.

When a process variable within a channel exceeds a predetermined extremum (setpoint), the associated bistable output relay will deenergize. Signals, cables, modules, dedicated power supplies and associated test circuitry are maintained independent across the four channels.

Contacts from the bistable relays of the same parameter in the four protective channels are arranged into six logic ANDS, designated AB, AC, AD, BC, BD, and CD, which represent all possible coincidences of two combinations. To form an AND circuit, the trip relay contacts associated with two like measurement channels are connected in parallel (e.g., one from A and one from B). This process is continued until all combinations have been formed.

Since there is more than one parameter that can initiate a reactor trip, the parallel pairs of trip relay contacts for each monitored parameter are connected in series (logic OR) to form six logic matrices, also designated AB, AC, AD, BC, BD, and CD.

Each matrix is powered from two diode isolated power supplies connected to two different vital power sources. Each power supply has dedicated isolation and ground fault detection circuitry.

Each logic matrix drives four matrix output relays. The contacts of the matrix output relays are combined into four trip paths, each trip path formed by connecting six contacts (one matrix output relay contact from each of the six logic matrices) in series. Each trip path is connected in series with an initiation relay which controls the power to the undervoltage and shunt trip coils of the reactor trip switchgear system (RTSS) circuit breakers. Four circuit breakers are provided. They are arranged in two parallel groups, consisting of two breakers in series in each group to control the power from two parallel motor-generator sets. Opening one breaker in each of the two groups will remove the power to both control element drive mechanism control system (CEDMCS) power supplies allowing all of the control element assemblies to drop into the core. Summarizing, coincident trip signals from two protective channels for the same trip parameter will scram the reactor.

In addition to the automatic trip of the reactor, means are also provided for a manual trip. Two independent sets of manual trip pushbuttons are provided to open the trip circuit breakers. Both manual trip pushbuttons in a set must be depressed to initiate a reactor trip.

7.2.1.1 Variable Overpower Trip

The variable overpower trip function has been provided CESSAR in lieu of the high linear power level trip provided for Arkansas Nuclear One, Unit 2 (ANO-2, NRC Docket 50-368) which was the first CE plant (prototype) to use this type of reactor protective system.

The circuitry employs solid state analogue and digital devices to initiate reactor trip if the rate of change of neutron power with time exceeds a predetermined setpoint. This feature provides protection for rapid power changes from initial power levels spanning zero to full power. The circuitry will also initiate a trip if the neutron power exceeds an absolute maximum value independent of its time rate of change.

The bistable circuit compares the process variable (neutron power) to a variable setpoint.

The variable setpoint follows the measured neutron power. At steady state the variable setpoint will be equal to the measured power. The time rate of change of the setpoint is electronically limited. Hence, the setpoint will lag the neutron power under transient conditions. When these signals differ by a predetermined amount the bistable will change state.

These features are testable. Special test equipment is provided as part of the CESSAR scope of supply.

7.2.1.2 Variable Low Reactor Coolant Flow Trip

The variable low reactor coolant flow circuitry provides protection for the hypothesized reactor coolant pump sheared shaft event. This feature was not provided for ANO-2.

Flow is sensed by measuring the pressure drop across the primary side of each steam generator (four measurement channels per steam generator).

A trip will be initiated should the time rate of change of primary coolant flow exceed a predetermined setpoint. The setpoint will be chosen such that the decrease of primary flow due to a RCP sheared shaft event will initiate a trip, while a pump coastdown on loss of power will not initiate a trip. The core protection calculators provide protection for the latter event. An absolute low flow trip is also provided as an integral feature of the circuitry. The variable overpower and low reactor coolant flow trips utilize the same circuitry described above.

7.2.1.3 High LPD/Low DNBR Trips

The core protection calculators (CPCs) and associated control element assembly calculators (CEACs) provide a reactor trip upon detection of high local power density (LPD) or low departure from nucleate boiling ratio (DNBR). This system is functionally identical to that employed at ANO-2. The hardware is different.

Significant hardware differences include: (1) procurement of the input/output chassis from ANALOGIC in lieu of SEL, (2) use of a newer model cotputer generating significantly less heat (Perkin Elmer 8/16 E in lieu of Perkin Elmer 7/16), (3) use of fiber optic data links between the CEACs and CPCs in lieu of optical couplers, and (4) deletion of digital data links between the CPCs and the plant computer/core monitoring computer. These changes should improve system availability and reliability.

The applicability of staff CPC concerns raised during the licensing of ANO-2 were reviewed. These issues were discussed at length at the independent design review (see Exhibit II.A.2 and associated transcript). Several concerns had been generically resolved based on technical analyses or imposition of administrative controls. Alternate resolutions and applicability of CESSAR to the concerns at ANO-2 is addressed below.

Isolation devices between qualified process variable instrumentation which provide input to the CPCs and nonqualified displays in the control room at ANO-2 have been deleted. The corresponding displays for CESSAR plants are qualified obviating the need for qualified isolation devices in this application. Where interfaces between the CPCs and non-IE equipment do exist qualified (and tested) isolation devices are employed.

Testing of electrical-magnetic interference (EMI) susceptibility was required at ANO-2. Generic EMI susceptibility studies have been performed for CESSAR. Reference plant specific testing/surveys will not be required. Deletion of this requirement is based on experience to date and the staff assessment that EMI susceptibility, should it occur at a reference plant, will be revealed during normal operation.

A six-month integrated burn-in test period was required for ANO-2 prior to plant operation. This test will not be required for CESSAR and/or reference plants based experience to date. Module burn-in testing and software testing will continue to be performed.

The digital data-links at ANO-2 between the CPCs and the plant computer have been deleted in the CESSAR design resolving this concern. (See Section 7.7.11.1, Remote Input System.)

The optical isolators at ANO-2 which link the two CEACs and the four CPCs are to be periodically tested. Fiber optics with a minimum distance of three feet are used in CESSAR in place of the optical isolators. The corresponding periodic resistance testing need not be performed.

Computer software modifications are to be performed in accordance with "CPC Protection Algorithm Software Change Procedure" CEN-39(A)-P, Revision 2 and Supplement 1-P Revision 01. This requirement is to be a condition of the license.

Software modifications to incorporate reactor power cutback have not been performed to date. These modifications will be audited as part of our on-going software verification and validation program.

Software modifications to incorporate plant-specific data constants are outside the CESSAR scope. (See Section 7.2.6 Setpoints.)

Issues related to periodic testing, changes to addressable constants, and field installation are outside the CESSAR scope. The installation and operational issues are to be addressed by the reference applicant.

7.2.2 Supplementary Protection System

The function of the SPS is to initiate a diverse reactor trip upon detection of high RCS pressurizer pressure. The SPS consists of four independent instrument channels. Each SPS channel initiation (output) relay when deenergized will inturn deenergize the undervoltage coil and energize the shunt coil of one of the four RTSS circuit breakers. Selective two of four reactor trip logic is performed by the RTSS circuit breaker configuration.

The RPS and SPS were designed and fabricated by different manufacturers using diverse circuits and components. The SPS power supplies are diverse and the output voltage unique. The SPS and RPS are physically separated. Pressure transducers for the SPS and RPS were obtained from different manufacturers. The process loop and bistable designs are different. The SPS contains a zero adjustment and a discrete plus continuous fine setpoint adjustment. The RPS bistable does not have an offset (zero) adjustment and uses a continuous set-point adjustment. Hence, maintenance calibration and adjustment by the instrument technician will be different.

Testing of the SPS is performed by summing the process loop signal and an injected test signal and observing RTSS circuit breaker trip. Testing may be performed on-line. The SPS uses dedicated test equipment. Testing is to be incorporated in the reference plant technical specifications.

One of the four RPS channels may be bypassed. The SPS contains no electrical bypasses. Should an SPS channel fail high, the corresponding RTSS circuit breaker will trip, and the RTSS will be in a selective 1 of 3 logic configuration. Should the SPS channel fail low, the RTSS will be in selective 2 of 3 logic configuration with respect to the SPS and selective 2 of 4 logic configuration with respect to the RPS high pressure trip.

Both systems sense RCS pressurizer pressure, employ solid state operational amplifiers, utilize deenergize to actuate output relays, and actuate the four RTSS circuit breakers.

The SPS conforms to the regulations, regulatory guides, and industry standards applicable to the RPS. Furthermore, the CESSAR scope of supply of the SPS meets practical achievable diversity, vis-a-vis the RPS, and hence in the staff's judgment meets the intent of NUREG-0460, Volume 3, Anticipated Transient Without Scram (ATWS). Refer to Section 15.3.9 for further discussion of this issue.

The Commission has not ruled on ATWS to date. Hence, the staff cannot form conclusions concerning the suitability of the SPS as a long-term ATWS solution. Installation of supplementary protective system is not a current regulatory requirement. Should the SPS be considered on ATWS solution, reference plants will be required as a license condition to maintain the component manufacturer diversity of the SPS and RPS over the life of the plant.

7.2.3 Equipment Protection Trips

Upon reactor trip the turbine is also tripped and main feedwater is run back to 5% of full flow. These features do not fully conform to the criteria applicable to the RPS. Operation of these trips has not been nor should be assumed in the safety analyses.

Conversely, reactor trip due to turbine trip is not provided in CESSAR (see 7.7, reactor power cutback system).

7.2.4 Diverse RTSS Testing

The reactor protective system (RPS), supplementary protective system (SPS), and manual trip, all deenergize the undervoltage coil and energize the shunt trip coil of each of the four circuit breakers of the reactor trip switchgear system. These diverse actuation circuits (undervoltage and shunt trip coils) are individually testable with the plant at power or shutdown.

Reference plants are to independently test these features at a minimum frequency of once each 18 months. This test requirement is to be incorporated in the technical specifications.

7.2.5 RPS Testing

The RPS includes test circuits and features which permit functional overlap testing with the plant at power.

Response time testing is to be addressed by the reference plant.

7.2.6 RPS Setpoints

Numerical values of setpoints are outside the CESSAR scope of supply and are to be included in reference plant technical specifications. The limiting safety system setpoints are to be calculated using the maximum transducer errors in the nonconservative direction observed during environmental testing of the transducer over the longest time period for which the specific equipment is intended to function. This longest time period may be different for different events analyzed. The setpoints are to include the effects of level measurement accuracy due to reference leg environmental exposure consistent with the concerns of IE Bulletin 79-21. Numerical values of level setpoints are to be selected such that, when accumulative errors are considered, the actual water level at which a level trip occurs will not be within 5% of the level measurement span. Deviations from this setpoint calculational methodology are to be addressed by reference applicants in their Safety Analysis Report. The above restrictions are to be included as license conditions for the reference plant.

7.2.7 RPS Bypass

7.2.7.1 Operational Bypasses

Operational bypasses are manually instated and interlocked with electrical permissive features and automatically removed. These features conform to the requirements of IEEE Standard 279.

7.2.7.2 RPS Channel Bypasses

Individual trip parameters may be electrically bypassed. Electrical interlocks are provided such that no more than one of the four channels of a given like trip parameter may be simultaneously bypassed. With a trip parameter in bypass the RPS will function as a 2 of 3 logic trip network with respect to the bypassed trip parameter and will continue to function as a 2 of 4 logical trip network with respect to the remaining non-bypassed trip parameters. The RPS while operating in this bypassed mode meets applicable criteria including IEEE Standards 279, 379, and 384. Hence, such operation is acceptable. Nevertheless, the staff believes that it is prudent that an inoperable channel should be repaired and returned to service as quickly as practicable. When a protection channel of a given process variable becomes inoperable, the defective channel may be placed in bypass until the next reference plant safety committee meeting at which time the reference plant safety committee will be required by the technical specifications to review and document their judgment concerning prolonged operation in bypass, channel trip, and/or repair. The goal should be to return the channel to its operable state as soon as practicable. In any case, the technical specifications will require any inoperable protection channel to be repaired and restored to an operable state upon obtaining the first cold shutdown operational mode following channel malfunction.

7.2.8 Evaluation Findings

We have conducted an audit review of the Reactor Protective System (RPS) to include the supplementary protective system for conformance to guidelines of the applicable regulatory guides and industry codes and standards. In Section 7.1 of this SER, we conclude that the applicant had adequately identified in CESSAR the guidelines applicable to these systems. Based upon our audit review of the design for conformance to these guidelines, we find that there is reasonable assurance that the CESSAR scope of supply of these systems will conform to the guidelines applicable to them.

Our review has included the identification of those systems and components for the RPS which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. Based upon our review, we conclude that the applicant has identified the systems and components consistent with the design bases for the RPS. Sections 3.10 and 3.11 of this SER address the qualification programs to demonstrate the capability of these systems and components to survive applicable events. Therefore, we find that the identification of the systems and components satisfies this aspect of the GDC-2, "Design Bases for Protection Against Natural Phenomena," and GDC-4, "Environmental and Missile Design Bases."

Based our review, we conclude that the RPS conforms to the design bases requirements of IEEE Standard 279. The RPS includes the provision to sense accident conditions and anticipated operational occurrences and initiate reactor shutdown consistent with the analysis presented in Chapter 15 of the CESSAR FSAR. Therefore, we find that the RPS satisfies the requirements of GDC-20, "Protection System Functions."

The RPS adequately conforms to the guidance for periodic testing in Regulatory Guide 1.22 and IEEE Standard 338 as supplemented by Regulatory Guide 1.118.

The bypassed and inoperable status indication adequately conforms to the guidance of Regulatory Guide 1.47.

The RPS adequately conforms to the guidance on the application of the single failure criterion in IEEE Standard 379, as supplemented by Regulatory Guide 1.53. Based on our review, we conclude that the RPS satisfies the requirement of IEEE Standard 279 with regards to system reliability and testability. Therefore, we find that the RPS satisfies the requirement of GDC-21, "Protection System Reliability and Testability."

The RPS adequately conforms to the guidance in IEEE Standard 384 as supplemented by Regulatory Guide 1.75 for the protection system independence. Based on our review, we conclude that the RPS satisfies the requirement of IEEE Standard 279 with regards to the independence of systems. Therefore, we find that the RPS satisfies the requirement of GDC-22, "Protection System Independence."

Based on our review of failure modes and effects for the RPS, we conclude that the system is designed to fail into a safe mode if conditions such as disconnection of the system, loss of energy, or a postulated adverse environment are experienced. Therefore, we find that the RPS satisfies the requirements of GDC-23, "Protection System Failure Modes."

Based on our review of the interfaces between the RPS and plant operating control systems, we conclude that the system satisfies the requirements of GDC-24, "Separation of Protection and Control Systems."

Based on our review of the Reactor Protective System, we conclude that the system satisfies the protection system requirements for malfunctions of the reactivity control system, such as accidential withdrawal of control rods. Section 15 of the SAR addresses the capability of the system to assure that fuel design limits are not exceeded for such events. Therefore, we find that the RPS satisfies the requirements of GDC-25, "Protection System Requirements for Reactivity Malfunction." Our conclusions, noted above, are based on the requirements of IEEE Standard 279 with respect to the design of the RPS. Therefore, we find that the RPS satisfies the requirement of 50.55a(h) with regards to IEEE Standard 279.

Our review of the RPS has examined the dependence of this system on the availability of essential auxiliary support (EAS) systems. Based on our review, we conclude that the design of the RPS is compatible with the functional performance requirements of EAS systems. Therefore, we find the interfaces between the RPS design and the design interfaces with the EAS systems to be acceptable. In summary, the staff concludes that the design of the Reactor Protective System (RPS) and the design of the essential auxiliary support (EAS) systems are acceptable and meet the relevant requirements of GDC 2, 4, 20, 21, 22, 23, 24, and 25, and 10 CFR Part 50.55a(h).

7.3 Engineered Safety Features Actuation System

The CESSAR scope of supply includes selected transducers, process electronics, bistables, logic, ESFAS auxiliary relay cabinet actuation relays, and selected actuated devices. Staff review of all equipment upsteam of the process electronics and downsteam of the ESFAS auxiliary relay cabinet will be performed on a plant-specific basis. CE provides design interface requirements for in-scope and out-of-scope equipment as described in Section 7.1.

7.3.1 System Description

The engineered safety features actuation system (ESFAS) is part of the overall plant protection system provided by Combustion Egineering. The Reactor Protection System and the supplementary protective system, discussed in Section 7.2, form the balance of the plant protection system.

The engineered safety features are designed to mitigate the consequences of postulated design basis events. The engineered safety features function to limit, contain, control, and terminate postulated event sequences.

The ESFAS monitors selected parameters and generates appropriate signals whenever established limits are reached. The signals generated and the associated trip parameters are as follows:

(1) Safety injection actuation signal (SIAS) - low pressurizer pressure or high containment pressure.

(2) Recirculation actuation signal (RAS) - low refueling tank water level.

(3) Containment spray actuation signal (CSAS) - high-high containment pressure

(4) Containment isolation actuation signal (CIAS) - same as SIAS

(5) Main steam isolation signal (MSIS) - low steam generator pressure or high steam generator level in either steam generator or high containment pressure.

(6) Emergency feedwater actuation signal to steam generator A (EFAS-1) - low steam generator A water level coincident with not low differential pressure of steam generator A relative to steam generator B.

(7) Emergency feedwater actuation signal to steam generator B (EFAS-2) - low steam generator B water level coincident with not low differential pressure of steam generator B relative to steam generator A.

Each of the trip parameters listed above is monitored by four independent measurement channels. The signals generated by the four measurement channels are received by four trip bistables. At the bistables the signals are compared to predetermined setpoints. Whenever a channel parameter reaches the predetermined setpoint, the bistable initiates a channel trip. The bistable output signals feed two-out-of-four coincidence logic matrices.

The outputs of the logic matrices feed the ESFAS auxiliary relay cabinet actuation relays, which in turn initiate the redundant engineered safety features and supporting equipment. Manual initiation is also provided. Deenergize to actuate relay logic is consistently used.

Once initiated the ESFAS auxiliary relay cabinet actuation relays will remain in their emergency state until manually reset. Electrical interlocks prevent ESFAS reset until the process parameter which initiated the ESFAS is within normal bounds. Reset time delay interlocks are not employed.

Test features and test equipment are provided to permit overlap testing from process electronics to group actuation relays at power. Interface documents assign actuated devices to group actuation relays. These group assignments permit full system testing. For example, given a pump and valve in series and assigned to different group actuation relays, the pump may be started by a test with the valve closed, and the valve stroked in a subsequent test with the pump not running.

7.3.2 EFAS

The function of the emergency feedwater actuation signal (EFAS) is to intiate feedwater flow to each of the steam generators upon detection of steam generator low level. Flow to a given steam generator is initiated if and only if that steam generator has not depressurized relative to the other steam generator.

The emergency feedwater actuation signals ultimately start pumps and open valves. Once initiated the group actuation relays assigned to start pumps will stay in their emergency state until manually reset. In contrast the group actuation relays used to control valves will cycle based on steam generator level detected by the wide range steam generator level instrumentation. As other ESFAS signals, EFAS employs selective 2 of 4 actuation logic. One channel of a given process variable may be bypassed. Electrical interlocks prohibit bypass of more than one channel. Unlike other ESFAS signals which employ logical OR circuitry exclusively for diverse process variables, EFAS employs in part logical AND circuitry. Hence, the interrelationhip of bypass of single channels of diverse process variables must be considered. Electrical interlocks to prevent bypass of more than one channel of diverse variables is not provided. Reference plants are to adopt technical specifications which limit those multiple bypasses which would result in the remaining operable channels not meeting the single failure criterion.

Individual reference plants may choose to operate selected emergency feedwater valves upon generation of EFAS (cycling) and MSIS (close). Control logic provided by the reference plant is to be configured such that EFAS overrides MSIS.

EFAS is part of the engineered safety features actuation system and hence, meets the requirements of Task Action Plan Item II.E.1.2 (NUREG-0737).

7.3.3 Low Pressure Bistables

Low pressurizer pressure and low steam generator pressure bistables include features which permit manual reduction of their setpoints by predetermined increments below the process variable value. This feature is used during plant cooldowns to prevent inadvertent SIAS and MSIS. Should the RCS or steam generator pressures be increased following manual decrease of the setpoint, the more conservative higher setpoint would be automatically reinstated. These features are employed in lieu of simple operational bypass of the protective function and provide continuous protection during cooldown should the pressure decrease below the reduced setpoint.

7.3.4 Reference Plant Features

Bypass and inoperable status indication (Regulatory Guide 1.47), status indication, manual initiation and ESF control systems, and overcurrent protection, for actuated equipment are outside the CESSAR scope of supply. Interface documents are provided as described in Section 7.1.

7.3.5 ESFAS Channel Bypasses

Staff evaluation of RPS channel bypasses, Section 7.2.7.2, is applicable to the ESFAS. Reiterating, operation with a channel in bypass is permissible and the system when operating with a channel in bypass meets applicable regulatory criteria. The staff believes that it is prudent to restrict the time that channels are in bypass (or trip) and will include restrictions in the reference plant technical specifications to limit such operation.

7.3.6 ESFAS Setpoints

Numerical values of setpoints are outside the CESSAR scope of supply and are to be included in the reference plant technical specifications.

Evaluation of RPS setpoint methodology (Section 7.2.6) is equally applicable to ESFAS setpoints.

7.3.7 Evaluation Findings

The review of the instrumentation and control aspects of the engineered safety feature (ESF) systems was restricted to the engineered safety features actuation system (ESFAS) and excluded ESF actuated devices and ESF control systems which regulate the operation of ESF systems. The ESFAS detects a plant condition requiring the operation of an ESF system and/or essential auxiliary support (EAS) system and initiates operation of these systems.

We have conducted an audit review of the ESFAS for conformance to guidelines of the applicable regulatory guides and industry codes and standards. In Section 7.1 of this SER, we concluded that the applicant had adequately identified in CESSAR the guidelines applicable to these systems. Based upon our audit review of the system design for conformance to the guidelines, we find that there is reasonable assurance that the systems conform to the applicable guidelines.

Our review has included the identification of those systems and components for the ESFAS which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. Based upon our review, we conclude that the applicant has identified those systems and components consistent with the design bases for the systems. Sections 3.10 and 3.11 of this SER address the qualification programs to demonstrate the capability of the systems and components to survive applicable events. Therefore, we find that the identification of the systems and components satisfies this aspect of the GDC-2, "Design Bases for Protection Against Natural Phenomena," and GDC-4, "Environmental and Missile Design Bases."

Based on our review, we conclude that the ESFAS conforms to the design bases requirements of IEEE Standard 279. The system includes the provisions to sense accident conditions and anticipated operational occurrences to initiate the operation of ESFAS consistent with the analyses presented in Chapter 15 of the CESSAR FSAR. Therefore, we find that the ESFAS satisfies the requirents of GDC-20, "Protection System Functions."

The ESFAS adequately conforms to the guidance for periodic testing in Regulatory Guide 1.22 and IEEE Standard 338 as supplemented by Regulatory Guide 1.118. The bypassed and inoperable status indication of the ESFAS adequately conforms to the guidance of Regulatory Guide 1.47. The ESFAS adequately conforms to the guidance on the application of the single failure criterion in IEEE Standard 379 as supplemented by Regulatory Guide 1.53. Based on our review we conclude that the ESFAS satisfies the requirement of IEEE Standard 279 with regards to the system reliability and testability. Therefore, we find that the ESFAS satisfies the requirement of GDC-21, "Protection System Reliability and Testability."

The ESFAS adequately conforms to the guidance in IEEE Standard 384 as supplemented by RG 1.75 for the protection system independence. Based on our review, we conclude that the ESFAS satisfies the requirement of IEEE Standard 279 with regards to the systems independence. Therefore, we find that the ESFAS satisfies the requirement of GDC-22, "Protection System Independence."

Based on our review of the ESFAS, we conclude that the system is designed with due consideration of safe failure modes if conditions such as disconnection of the system, loss of energy, or a postulated adverse environment are experienced. Therefore, we find that the ESFAS satisfies the requirements of GDC-23, "Protection System Failure Modes."

The ESFAS and plant operating control systems do not share common components. Hence, we conclude that the system satisfies the requirements of IEEE Standard 279 with regards to control and protection system interactions. Therefore, we find that the ESFAS satisfies the requirement of GDC-24, "Separation of Protection and Control Systems."

Our conclusions noted above are based on the requirements of IEEE Standard 279 with respect to the design of the ESFAS. Therefore, we find that the ESFAS satisfies the requirement of 50.55a(h) with regards to IEEE Standard 279. Our review of the ESFAS has examined the dependence of these systems on the availability of essential auxiliary supporting (EAS) systems. Based on our review and coordination with those having primary review responsibility of the EAS systems, we conclude that the design interfaces with the ESFAS are compatible with the functional performance requirements of EAS systems. Therefore, we find the interfaces between the ESFAS and the design interfaces with the EAS systems to be acceptable.

In summary, the staff concludes that the ESFAS design is acceptable and meets the relevant requirements of GDC 2, 4, 20 thru 24, and 10 CFR Part 50.55a(h).

7.4 Systems Required For Safe Shutdown

CESSAR scope of supply includes: (1) instrumentation and controls described in Section 7.5, Safety Related Display Instrumentations which are used during normal operation and safe shutdown, (2) emergency feedwater actuation discussed in Section 7.3, ESFAS, (3) instrumentation and controls associated with the shutdown cooling system and the chemical volume control systems, discussed below.

Interface documents are provided for in-scope equipment and out-of-scope equipment which interface directly with the CESSAR scope of supply (e.g., steam generator atmospheric dump valves) or function as an auxiliary or supporting system (e.g., electrical distribution systems) as described in Section 7.1.

7.4.1 Shutdown Cooling System

The shutdown cooling system (SCS) is manually initiated. Two 100% redundant SCS trains are provided. Isolation valves and interlocks are discussed in Section 7.6. Interface documents are provided requiring manual control capability and valve position indication. Safety-grade indication of SCS heat exhanger and pump performance (pressure, temperature, flow) for each train is provided by CESSAR. The shutdown cooling system logic diagram shown in CESSAR Figure 7.4-1 shows the relationship of pressure interlocks, manual control, and SIAS and RAS, to actuated equipment. Actuated equipment initiation circuitry, provided by the reference plant, implements the desired logic, employing permissive and ESFAS signals provided by CESSAR.

7.4.2 Chemical and Volume Control System

The function of the chemical volume control system (CVCS) with respect to safe shutdown is to provide a source of borated water for inventory and reactivity control. The CVCS logic diagram shown in CESSAR Figure 7.4-2 shows the relationship of automatic level, pressure and flow signals, and manual actions to operate RCP seal injection flow, auxiliary spray, and charging flow. Only charging flow is required for safe shutdown. Actuated equipment initiation circuitry, provided by the reference plant, implements the desired CESSAR logic in accordance with interface documents.

The charging pump control circuitry includes high pressurizer level, low suction pressure, and lube seal pressure pump trips. The CESSAR scope of supply includes these features and remotely located override of these equipment protection features.

7.4.3 Remote Shutdown Capability

Remote shutdown capability will be reviewed on each reference plant. Interface documents require remote installation of selected instruments and controls associated with CESSAR scope systems (CESSAR 7.4.1.1.10).

7.4.4 Capability for Safe Shutdown Following Loss of a Bus Supplying Power to Instrumentation and Controls

The staff requested that the applicants review the adequacy of emergency operating procedures to be used to obtain safe shutdown upon loss of any Class 1E or non-Class 1E bus supplying power to safety or nonsafety-related instruments and controls. This issue was addressed for operating reactors through IE Bulletin 79-27. This concern is to be addressed by each reference plant.

7.4.5 Evaluation Findings

The review of systems required for safe shutdown included the sensors, circuitry, and redundancy features within the CESSAR scope of supply that provide the instrumentation and control functions that prevent the reactor from returning to criticality and provide means for adequate decay heat removal.

We have conducted an audit review of these systems for conformance to guidelines of the applicable regulatory guides and industry codes and standards. In

Section 7.1 of this SER we concluded that the applicant had adequately identified in CESSAR the guidelines applicable to these systems. Based upon our audit review of the systems designs for conformance to the guidelines, we find that there is reasonable assurance that the systems conform to the applicable guidelines.

Our review has included the identification of those systems and components required for safe shutdown which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. Based on our review, we conclude that the applicant has identified those systems and components consistent with the design bases for the systems. Sections 3.10 and 3.11 of this SER address the qualification programs to demonstrate the capability of these systems and components to survive applicable events. Therefore, we find that the identification of these systems and components satisfies this aspect of the GDC-2, "Design Bases for Protection Against Natural Phenomena," and GDC-4, "Environmental and Missile Design Bases."

Our review of the instrumentation and controls required for safe shutdown has examined the dependence of those systems within the CESSAR scope of supply on the availability of essential auxiliary support (EAS) systems. Based on our review and coordination with those having primary review responsibility for the EAS systems, we conclude that the design interfaces of EAS systems are compatible with the functional performance requirements of the systems reviewed in this section. Therefore, we find the interfaces between the design of safe shutdown systems and the design interface requirements of EAS systems to be acceptable.

In summary, the staff concludes that the systems required for safe shutdown are acceptable and meet the relevant requirements of GDC 2, 4, 13.

7.5 Safety-Related Display Instrumentation

Safety-related display instrumentation within the CESSAR scope of supply has been classified in the following categories:

(a) safety-related plant process display instrumentation
(b) reactor trip system monitoring
(c) engineered safety feature system monitoring
(d) CEA position indication
(e) postaccident monitoring
(f) automatic bypass indication

Display instrumentation within the CESSAR scope provide information related to CESSAR scope systems, i.e., primary system, steam generator, charging system, safety inspection system, shutdown cooling system, chemical and volume control system including refueling water tank. Display instrumentation for essential auxiliary systems is provided by the reference plant.

7.5.1 Description

Safety-related process instruments are identified in Table 7.5-1 of CESSAR and were identified as safety grade at the CESSAR instrumentation and control independent design review. All direct process variables which initiate reactor trip or an ESFAS signal are indicated or recorded. Derived values of local power density and DNBR are also displayed. Reactor trip system monitoring includes indication of: (1) process parameters which initiate reactor trip, (2) trip, pretrip, and bypass lights, (3) audible alarms, (4) CEA dropped rod information, and (5) trip swtichgear position.

Engineered safety features monitoring instrumentation is shown in Table 7.5-2 of CESSAR. These instruments span the CESSAR scope of emergency safety features. Containment isolation and main steam and feedwater isolation valve position indication circuitry is provided by the reference plant as an interface requirement. The emergency feedwater system is not within the CESSAR scope of supply. EFW flow instrumentation is to be provided by the reference plant. In addition, four control modules of the ESFAS (Section 7.3) provide pretrip, trip, bypass, and operating bypass indication for each ESFAS signal.

7.5.2 Postaccident Monitoring

Postaccident monitoring instrumentation was discussed at length at the Independent Design Review meeting (Exhibit IIB-27 through IIB-45). CESSAR currently provides many of the instruments enumerated in Regulatory Guide 1.97, Rev. 2, "Instrumenation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980. A program is underway to identify, and upgrade if required, instrumentation needed to accomplish preplanned manual actions. Instrumentation to assess the magnitude of radioactive releases is to be provided by the reference plant.

The staff will perform a comprehensive review of postaccident monitoring instrumentation in conjunction with the review of reference plants.

7.5.3 Automatic Bypass Indication

CESSAR provides automatic bypass indication for the RPS and ESFAS as described in Section 7.5.1.

Automatic bypass indication for actuated equipment is provided by the reference plant.

7.5.4 Evaluation Findings

The information systems important to safety provide the operator with information on the status of the plant. The CESSAR scope of supply provides a subset of information needed by the operator to perform manual safety actions. The scope of review included tables of system variables and component states to be indicated, functional diagrams, electrical and physical layout drawings, and descriptive information. The review has included the applicable acceptance criteria and guidelines and design bases, including those for indication of bypassed or inoperable safety systems. The review has also included the CESSAR analyses of the manner in which the design of information systems conforms to the acceptance criteria and guidelines which are applicable to these systems as noted in the staff's Standard Review Plan.

We have conducted an audit review of these systems for conformance to guidelines of the applicable regulatory guides and industry codes and standards. In Section 7.1 of this report, we concluded that the applicant had adequately identified in CESSAR the guidelines applicable to these systems. Based upon our audit review of the systems designs for conformance to the guidelines, we find that there is reasonable assurance that the CESSAR scope of supply conforms to the guidelines applicable to them.

Our review has included the identification of those systems and components of the information systems which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. Based upon our review, we conclude that CESSAR has identified those systems and components consistent with the design bases for the systems. Sections 3.10 and 3.11 of this report address the qualification programs to demonstrate the capability of these systems and components to survive applicable events. Therefore, we find that the identification of these systems and components satisfies this aspect of GDC-2, "Design Bases for Protection Against Natural Phenomena," and GDC-4, "Environmental and Missile Design Bases."

The redundant safety-grade information systems within the CESSAR scope of supply adequately conform to the guidance for the physical independence of electrical systems provided in RG 1.75. We conclude that the information systems important to safety include appropriate variables for the CESSAR scope of supply and that their range and accuracy are consistent with the plant safety analysis. Therefore, we find that these information systems satisfy the requirements of GDC-13, "Instrumentation and Control," for monitoring variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions. Further, we find that conformance to GDC-13 and the applicable guidelines satisfies the requirements of GDC-19, "Control Room," with respect to information systems within the CESSAR scope of supply provided in the control room.

In summary, the staff concludes that the information systems important to safety within the CESSAR scope of supply are acceptable and meet the requirements of GDC 2, 4, 13, and 19.

7.6 Interlock Systems Important to Safety

Systems described in this section are those required for safety but not previously discussed in Sections 7.2 through 7.5. The CESSAR scope of supply includes three such systems: shutdown cooling systems interlocks, safety injection tank isolation valve interlocks, and the low temperature overpressure mitigating system.

7.6.1 Description

The function of the shutdown cooling system interlocks is to prevent inadvertent overpressurization of the shutdown cooling system. Safety-grade, redundant interlocks employing diverse sensors prevent valve opening of the three in-series valves on the SCS suction lines when RCS pressure is greater than 400 psia and automatically close the valves if the RCS pressure is above 500 psia. One interlock controls one of the three SCS suction line isolation valves, the other interlock controls the remaining two SCS isolation valves. The system is configured such that no single failure will preclude opening of at least one SCS path or of closing both paths.

The function of the safety injection tank (SIT) interlocks is to provide an automatic SIT isolation valve open command if RCS pressure is above 500 psig and to provide an open permissive signal if RCS pressure is less than 415 psig.

SIAS overrides this interlock and opens the SIT isolation valves. As noted in Section 7.3, the pressurizer pressure setpoint is reduced during plant cooldown and SIAS is not bypassed.

These interlocks employ common sensors and logic cards with multiple bistables. CESSAR provides interlock contacts. The reference plant will use these contacts in motor control circuitry, in accordance with interface documents, to effect the desired logic. These interface documents also require independent emergency power for the redundant interlocks and valve motor operators.

The function of the shutdown cooling system relief valves (mechanical code safety valves) is to prevent inadvertent overpressurization of the RCS at low temperature. The setpoint of these redundant mechanical valves is 435 psig. The valves are sized such that pressure overshoot will not result in RCS pressure reaching the SCS suction line isolation valve automatic close interlock setpoint of 500 psia. The shutdown cooling relief valves are on the SCS side of the SCS isolation valves. Alarms to indicate to the operator that a low temperature overpressure event is in progress or to indicate the shutdown cooling relief valves should be armed by virtue of manual opening of the SCS isolation valves is not within the current CESSAR scope of supply. These alarms should be provided by reference plants.

7.6.2 Evaluation Findings

The staff concludes that the designs of the interlock systems important to safety within the CESSAR scope of supply are acceptable and meet the relevant requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and 4, "Environmental and Missile Design Bases." This conclusion is based on the following.

The review of the interlock systems important to safety included the interlocks to prevent overpressurization of low pressure systems when connected to the primary coolant system. The staff position with regard to this interlock system is set forth in Branch Technical Position ICSB-3, "Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System." Based on our review, we conclude that the design of this system adequately complies with the staff's guidance.

Our review included the interlock provided to prevent overpressurization of the primary coolant system during low temperature operation. The staff's position with regards to this interlock system is set forth in Branch Technical Position RSB5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating At Low Temperatures." Based on our review, we conclude that the CESSAR scope of supply of this system adequately complies with the staff's guidance.

Our review included the interlocks for the safety injection tank valves. The staff's position with regards to this interlock system is set forth in Branch Technical Position ICSB-4, "Requirements of Motor Operated Valves in the ECCS Accumulator Lines." Based on our review, we conclude that these interlocks adequately comply with the staff's guidance.

Based on our review of the interlock systems important to safety, we conclude that their design bases are consistent with the plant safety analysis and the systems importance to safety. Further, we conclude that the aspects of the design of these systems with respect to single failures, redundancy, independence, qualification, and testability are adequate to assure that the functional performance requirements will be met.

Our review has included the identification of those systems and components of interlock systems important to safety which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. Based upon our review, we conclude that CESSAR has identified the systems and components consistent with the design bases for the interlock systems. Sections 3.10 and 3.11 of this SER address the qualification programs to demonstrate the capability of these systems and components to survive applicable events. Therefore, we find that the identification of the systems and components satisfies this aspect of the GDC-2, "Design Bases for Protection Against Natural Phenomena," and GDC-4, "Environmental and Missile Design Bases."

7.7 Control Systems

The plant control and monitoring system described in this section include the following:

(1) Reactivity control systems
(2) Reactor coolant system pressure control system
(3) Pressurizer level control system
(4) Feedwater control system
(5) Steam bypass control system
(6) Reactor power cutback system
(7) Boron control system
(8) Incore instrumentation system
(9) Excore neutron flux monitoring system
(10) Core operating limit supervisory system
(11) Plant monitoring system

7.7.1 Reactor Regulating System

The function of the reactor regulating system (RRS) is to automatically adjust reactor power and reactor coolant temperature to turbine demand. The RRS receives a turbine load index signal, reactor coolant hot and cold leg temperatures, and power range neutron flux. The RRS issues demand signals to the control element drive mechanism control system (CEDMCS) to: (1) drive CEAs at a high or low rate, (2) drive CEAs in or out of the core or hold them. An automatic withdrawal prohibit is also passed to the CEDMCS if RCS average temperature exceeds a reference temperature by a predetermined amount.

There are four rod control modes, sequential group movement in manual or automatic control, manual group movement, and manual individual CEA group movement. Sequential group movement in accordance with predetermined group overlap is controlled by signals received from the plant monitoring system (PMS). The PMS monitors rod positions utilizing signals from the CEDMCS up-down pulse counters, i.e., CEA demand position.

7.7.2 Reactor Coolant System Pressure Control System

The function of the pressurizer pressure control systems (PPCS) is to maintain RCS pressure within limits by controlling pressurizer heaters and spray valves. Proportional controllers or manual controls are employed. Low pressurizer level and high pressurizer pressure interlocks are also provided to shut off the pressurizer heaters.

7.7.3 Pressurizer Level Control System

The function of the pressurizer level control system (PLCS) is to maintain pressurizer level which is programmed as a function of RCS average temperature. This system controls charging pump operation and letdown flow rate. Manual control is also provided.

There are three charging pumps, two normally running. One charging pump is always running. If pressurizer level is abnormally high the PLCS will deenergize one normally running charging pump. If pressurizer level is abnormally low the PLCS will start the third, standby charging pump.

CESSAR provides PLCS high-level contacts which are employed by the reference plant in charging pump motor control circuits. Since the charging pumps are needed for safe shutdown and the PLCS is a control-grade system, reference plants should provide the capability to defeat the PLCS high-level interlock from the control room.

7.7.4 Feedwater Control System

The function of the feedwater control system (FWCS) is to automatically control steam generator level. A steam flow signal is used to control steam generator downcomer valve position. Steam generator level, compensated by the difference between feedwater and steam flow, is used to control economizer valve position. The higher of demand signals from the FWCS of either steam generator is used to control feedwater pump speed. Manual controls are provided. The system is designed to function between 15% and 100% power in the automatic mode. Below 15% power the system is manually operated.

7.7.5 Steam Bypass Control System

The function of the steam bypass control system (SBCS) is to control turbine bypass valves to dump steam to the condenser. Modulate and quick opening features are provided. Programmed main steam header pressure as a function of measured steam flow is compared to measured main steam header pressure, and biased by pressurizer pressure, to generate a modulate signal. The rate of change of an error signal of steam flow and pressurizer pressure is used to generate a quick open signal. Redundant circuitry is used to generate valve open permissive signals. Separate transmitters are employed and separate power supplies are required as a design interface. The modulate and quick open signals, and the redundant permissive signals function as a two-out-of-two control system, i.e., two random failures are required to inadvertently open a turbine bypass valve.

Should the turbine bypass valves be unable to meet steam dump demand, redundant electrical demand signals are sent from the SBCS to the reactor power cutback system.

7.7.6 Reactor Power Cutback System

The function of the reactor power cutback system (RPCS) is to provide a step change in reactor power following a large turbine load rejection, turbine trip, or loss of one of the two main feedwater pumps. The RPCS is a microprocessor-based system receiving redundant signals from the SBCS and feedwater pumps. The RPCS sends redundant signals to CEDMCS to drop the first or first and second control banks. Bank drop selection is determined by the plant monitoring system based on CEDM position and calculated power defect as a function of fuel cycle exposure. These signals are passed from the PMS to the RPCS.

The RPCS issues redundant commands to CEDMCS for all 24 CEA subgroups. Hard-wired jumpers in the CEDMCS prevent transmittal of these command signals to all but the first three CEA subgroups which comprise the first and second control banks which are to be inserted in the core. This feature prevents credible faults of the RPCS resulting in inadvertent subgroup drops of other than the three selected subgroups.

Implementation of the RPCS will require programming changes of the CPCs, such that the CPCs will not recognize intentional RPCS operation as an inadvertent rod drop or misalignment (see 7.2.1.3 and 4.4.5).

7.7.7 Boron Control System

The function of the boron control system is to maintain long-term reactivity control. The boron concentration may be automatically or manually maintained in the chemical and volume control system volume control tank at a prescribed concentration using this system. The volume control tank is the normal source for the charging pumps.

A boronometer is used to detect the RCS boron concentration and control the volume control tank boron concentration. Processed signals are sent to the control room display, to a control room annunciator, and to the PMS.

7.7.8 Incore Instrumentation System

The function of the incore instrumentation system is to monitor the core power distribution. (See 7.7.10, COLSS.)

Sixty-one incore monitoring assemblies are located in fixed locations in the core. Each assembly has five axially distributed rhodium detectors. Signals are processed by an incore amplifier system and multiplexed using a remote input system to the PMS (see 7.7.11). The remote input system is physically located in the auxiliary protective cabinets which house the CPCs. Two movable incore detectors controlled by the PMS are also provided. The detectors may be located in any core location and provide the capability for full incore mapping.

The fixed incore instrumentation provides the ability to take a "snap shot" of the core power distribution. The movable detectors provide finer resolution and the capability to perform cross-calibration.

7.7.9 Excore Neutron Flux Monitoring System

Two control channels using uncompensated ion chambers provide neutron flux information to the operator and input to the RRS (7.7.1). These instruments are independent of the safety channels which provide input to the RPS (7.2). Two startup channels using BF_3 proportional counters are also provided for monitoring source level neutron flux. These channels provide information to the operator and do not perform an automatic control or safety function.

7.7.10 Core Operating Limit Supervisory System

The function of the core operating limit supervisor system (COLSS) is to monitor and display information to the operator such that the operator is assisted in maintaining the plant within limiting conditions for operation defined in the plant technical specifications. Control board indication of the following COLSS parameters is continuously available to the operator.

(1) Core power operating limit based on peak linear heat rate
(2) Core power operating limit based on margin to DNB
(3) Total core power
(4) Margin between core power and nearest core power operating limit
(5) Axial shape index

COLSS alarms are initiated if: (1) core power exceeds a core power operating limit, (2) azimuthal flux tilt exceeds azimuthal flux tilt limit.

COLSS algorithms calculate: (1) reactor coolant volumetric flowrate based on RCP pump speed and differential pressure, (2) reactor coolant ΔT power, (3) turbine power based on turbine first stage pressure, (4) calorimetric power based on a secondary side heat balance, (5) core power distributions based on incore detector signals and predetermined local peaking factors adjusted for CEA position, (6) peak linear heat rate, (7) margin to DNBR.

Offline testing capability is provided to insure proper execution of COLSS on the PMS.

7.7.11 Plant Monitoring System

The plant monitoring system consists of two general-purpose computers, a plant monitoring computer and a core monitoring computer, and a data aquisition system including a remote input system described below.

COLSS may be executed on either computer.

The PMS is used to monitor operation within the power dependent insertion limits (PDIL). The PMS monitors CEA position by counting demand pulses and drives CEA position displays. PDIL and per-PDIL, out of sequence, and CEA deviation alarms are provided.

The RMS also provides normal CEA control limits and issues rod motion commands to CEDMCS, and controls and processes data from the movable incore system.

The PMS also performs equipment monitoring and data logging functions.

7.7.11.1 Remote Input System

The remote input system (RIS) consists of four separate input processing computers mounted in the auxiliary protective cabinets (one per channel). Each channel processes up to 128 analogue to digital converters. Data is stored in each RIS memory and transmitted to the general-purpose computers upon demand. A common data highway connects the four RIS channels. The RIS functions as a qualified isolation device.

The digital data links between the CPCs and the PMS which existed in earlier designs is not employed in CESSAR (7.2.1.3). In the CESSAR design the CPC provides three inputs to the PMS: CPC calculated core power, linear heat rate, and DNBR. This information is stored in the CPC memory, converted to an analogue signal in the CPC Input/Output chassis, transmitted to the RIS, converted back to a digital signal by the RIS and stored, and transmitted to the plant/core monitoring computers upon demand.

7.7.12 Control System Failures

CE has been requested to perform studies of (1) the effects of consequential control system failures due to high energy line breaks, and (2) effects of single failures of components shared by control systems such as power supplies, transducers, and impulse lines. These studies involve equipment provided by CESSAR and the reference plant and, hence, are not amenable to generic efforts. Each reference plant will be required to address these issues.

7.7.13 Evaluation Findings

The control systems used for normal operation that are not relied upon to perform safety functions, but which control plant processes having a significant impact on plant safety, have been reviewed. These control systems include the reactivity control systems and the control systems for the primary and secondary coolant systems. The staff concludes that the control systems are acceptable and meet the relevant requirements of General Design Criteria 13, "Instrumentation and Control," and 19, "Control Room." This conclusion is based on the following:

Based on our review of the CESSAR design bases, functional diagrams, and discussion of the control systems presented in CESSAR, we conclude that the control systems are capable of maintaining system variables within prescribed operating limits. Therefore, we find that the control systems satisfy this aspect of GDC-13, "Instrumentation and Control."

Our review of control systems included the features of these systems for both manual and automatic control of the process systems. We find that the control systems permit actions which can be taken to operate the plant safely during normal operation, including anticipated operational occurrences; therefore, the control systems satisfy GDC-19, "Control Room," with regards to normal plant operations. The conclusions of the analysis of anticipated operational occurrences and accidents, as presented in Chapter 15 of the CESSAR FSAR, have been used to confirm that plant safety is not dependent upon the response of the control systems. We find that the control systems are not relied upon to assure plant safety and are, therefore, acceptable.

8 ELECTRIC POWER SYSTEMS

8.1 General

The design of the offsite power and onsite power systems is outside the design scope of CESSAR System 80 and will be evaluated in applications referencing CESSAR. We have identified in Table 8.1 of this report the interface acceptance criteria for the offsite and onsite power systems. These criteria will form the basis for our review of each application which incorporates the CESSAR design to determine overall design conformance with the Commission's requirements.

The acceptability and completeness of the power supply interface requirements for specific CESSAR systems are addressed in the applicable sections of this report. We have, however, identified the following general interface requirements for power systems that must be satisfied by each applicant referencing CESSAR.

8.2 Offsite Power System

We require that the reference plant design for the offsite power system shall satisfy the following interface requirements:

1. We require that the engineered safety features loads should normally be fed either directly from the offsite power system or from the main generator unit.

2. The CESSAR design does not provide for the disconnection of the reactor coolant pumps from the electric system in the event of frequency decay condition in the grid. CE has included an interface requirement in the CESSAR of 3 Hz/second for the limiting underfrequency decay rate. The consequences of frequency decays of up to 3 Hz/second (with bus voltage at its nominal value and with all reactor coolant pumps connected to their buses) on the reactor coolant system flow are not more severe than the consequences of loss of flow of the four reactor coolant pumps due to loss of power. For applications referencing CESSAR, where credit is taken for the reactor coolant pump coast down, we will require that the applicant either demonstrate that the effects of electric grid disturbances on his plant are such that the limiting underfrequency decay rate of 3 Hz/second is not exceeded, or the pumps must be disconnected on grid frequency excursions beyond the acceptable limits.

8.3 Onsite Power System

8.3.1 Alternating Current Power System

We require that the reference plant design for the onsite alternating current power system shall satisfy the following requirements:

1. Two redundant and independent standby alternating current power generators shall be provided to conform with the required redundancy of safety-related systems and components included in the CESSAR design. These standby

generators shall be designed to attain rated voltage and speed within 12 seconds following either a loss of offsite power to the ESF bus, or initiation of a Safety Injection Actuation System (SIAS) signal.

2. If the offsite power is lost at some time after the standby generators are up to rated voltage and speed, and after the required ESF equipment is running following SIAS, the following requirements shall be met:

   a. Interrupted ECCS flow to the core shall be fully reestablished within 13 seconds.

   b. Interrupted emergency feedwater flow to the steam generator(s) shall be fully reestablished within 15 seconds.

3. Four physically and electrically independent 120 volt, 60 hz, single phase, ungrounded vital instrument sources are required to provide power to CESSAR instrumentation used for protection. The output frequency shall be 60 ± .05 Ha and the output voltage shall be regulated to within ± 2% at full output for a load power factor greater than 0.8.

4. The onsite alternating current power system design shall satisfy all the criteria identified in Table 8-1.

8.3.2 Direct Current Power Systems

We require that the balance-of-plant design for the onsite direct current power system shall satisfy the following interface requirements:

1. Four independent batteries and battery chargers shall be provided to meet the power requirements for CESSAR System 80 design scope safety-related systems and components.

2. The onsite direct power system design shall satisfy all the criteria identified in Table 8-1.

Conclusion

We have reviewed the interface requirements imposed on the electric power system by CESSAR System 80 design. In specific applications referencing CESSAR, with the balance of plant design for the offsite power system, onsite alternating current power system, and onsite direct current power system satisfying the interface requirements cited above, there is reasonable assurance that the assumptions made in the CESSAR accident analysis with regard to safety system functions can be sustained and the applications referencing CESSAR will satisfy the applicable regulatory requirements, as shown in Table 8-1.

TABLE 8-1

INTERFACE ACCEPTANCE CRITERIA AND GUIDELINES FOR ELECTRIC POWER SYSTEMS

The matrix of Table 8-1 identifies the acceptance criteria (denoted by "A") and the guidelines (denoted by "G") and their applicability to the various sections of Chapter 8.0. The acceptance criteria define the requirements established by the Commission for power systems important to safety; the guidelines amplify these requirements and provide more explicit basis for evaluation of the conformance of the power systems to these Commission requirements.

CRITERIA	TITLE	APPLICABILITY (SAR SECTION) 8.2	8.3.1	8.3.2	REMARKS
1. General Design Criteria (GDC), Appendix A to 10 CFR Part 50					
a. GDC 2	Design Bases for Protection Against Natural Phenomena		A	A	
b. GDC 4	Environmental and Missile Design Bases		A	A	
c. GDC 5	Sharing of Structures, Systems, and Components	A	A	A	
d. GDC 17	Electric Power Systems	A	A	A	
e. GDC 18	Inspection and Testing of Electrical Power Systems	A	A	A	
f. GDC 50	Containment Design Bases		A	A	

TABLE 8-1 (Continued)

	CRITERIA	TITLE	APPLICABILITY (SAR SECTION) 8.2	8.3.1	8.3.2	REMARKS
2.	Regulatory Guides (RG)					
	a. RG 1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems		G	G	
	b. RG 1.9	Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants		G		See IEEE 387
	c. RG 1.32	Use of IEEE Std 308, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations"	G	G	G	See IEEE 308
	d. RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	G	G	G	
	e. RG 1.63	Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants		G	G	See IEEE 317
	f. RG 1.75	Physical Independence of Electric Systems		G	G	See IEEE 384
	g. RG 1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	G	G	G	

TABLE 8-1 (Continued)

CRITERIA	TITLE	APPLICABILITY (SAR SECTION) 8.2	8.3.1	8.3.2	REMARKS
h. RG 1.106	Thermal Overload Protection for		G	G	
i. RG 1.108	Periodic Testing of Diesel Generators		G		
j. RG 1.118	Periodic Testing of Electric Power and Protection Systems		G	G	See IEEE 338
k. RG 1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants			G	See IEEE 484
l. RG 1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants			G	See IEEE 450
3. Branch Technical Positions					
a. BTP ICSB 4	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines		G		
b. BTP ICSB 8 (PSB)	Use of Diesel-Generator Sets for Peaking		G		
c. BTP ICSB 11 (PSB)	Stability of Offsite Power Systems	G			
d. BTP ICSB 18 (PSB)	Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves		G		

TABLE 8-1 (Continued)

CRITERIA	TITLE	APPLICABILITY (SAR SECTION) 8.2	8.3.1	8.3.2	REMARKS
e. BTP ICSB 21	Guidance for Application of RG 1.47	G	G	G	
f. BTP PSB-1	Adequacy of Station Electric Distribution System Voltages		G		
h. BTP PSB-2	Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status		G		
4. NUREG Reports					
a. NUREG/ CR 0660	Enhancement of Onsite Diesel Generator Reliability		G		

9 AUXILIARY SYSTEMS

With the exception of the chemical and volume control system (CVCS) and portions of the fuel handling system, the plant auxiliary systems are the responsibility of applicants that reference CESSAR. CESSAR includes interface requirements for various auxiliary systems (including fuel storage, water systems, process auxiliaries, and heating, ventilation, and air conditioning systems) required for assuring a safe plant shutdown and for assuring the safety of the fuel storage facility in order to prevent unacceptable radiological releases to the environment. We have reviewed the CVCS and portions of the fuel handling system within the CESSAR scope and the interfaces identified for other auxiliary systems.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

CESSAR indicates that the design of the new fuel storage facility is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Sections 4.2.5 and 9.1.4 concerning the new fuel storage facility design and safety in order to assure that the CESSAR fuel handling accident analysis is bounding for CESSAR reference plants, thereby protecting against unacceptable radiological releases to the environment (see Section 15.4.6 of this report). These interfaces are in accordance with the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," 4, "Environmental and Missile Bases," 61, "Fuel Storage and Handling and Radioactivity Control," and 62, "Prevention of Criticality in Fuel Storage and Handling."

Based on our review of the CESSAR interfaces, we conclude that CESSAR provides adequate information relating to new fuel storage in order that referencing applicants can comply with the requirements of General Design Criteria 2, 4, 61, and 62 and is, therefore, acceptable and complete in this regard, except as noted in Section 15.4.6.

9.1.2 Spent Fuel Storage

CESSAR indicates that the design of the spent fuel storage facility is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Sections 4.2.5 and 9.1.4 concerning the spent fuel storage facility design and safety in order to assure that the CESSAR fuel handling accident analysis is bounding for CESSAR reference plants, thereby protecting against unacceptable radiological releases to the environment (see Section 15.4.6 of this report). These interfaces are in accordance with the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," 4, "Environmental and Missile Design Bases," 61, "Fuel Storage and Handling and Radioactivity Control," 62, "Prevention of Criticality in Fuel Storage and Handling," and 62, "Monitoring Fuel and Waste Storage," and the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," and 1.29, "Seismic Design Classification."

Based on our review of the CESSAR interfaces, we conclude that CESSAR provides adequate information relating to the spent fuel pool cooling system in order that referencing applicants can comply with the requirements of General Design Criteria 44, 61, and 63 and the guidelines of Regulatory Guide 1.13 and is, therefore, acceptable and complete in this regard, except as noted in Section 15.4.6.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

CESSAR indicates that the design of the spent fuel pool cooling and cleanup system is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Sections 9.1.4.6 and 9.3.4 to assure adequate spent fuel cooling and that the CESSAR fuel handling accident is bounding for CESSAR reference plants, thereby protecting against unacceptable radiological releases to the environment (see Section 15.4.6 of this report). These interfaces are in accordance with the requirements of GDC 44, "Cooling Water," 61, "Fuel Storage and Handling and Radioactivity Control," and 63, "Monitoring Fuel and Water Storage" and the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis." An interface is also included for using the spent fuel pool as an alternate source of water for boron injection through the Chemical and Volume Control System (see Section 9.3.4 of this report).

Based on our review of the CESSAR interfaces, we conclude that CESSAR provides adequate information relating to the spent fuel pool cooling system in order that referencing applicants can comply with the requirements of GDC 44, 61, 63 and the guidelines of Regulatory Guide 1.13 and is, therefore, acceptable and complete in this regard, except as noted in Section 15.4.6.

9.1.4 Fuel Handling System

The fuel handling system, in conjunction with the fuel storage area, provides a means of transporting, handling, and storing of fuel. The fuel handling system consists of equipment necessary for the safe handling of new and spent fuel assemblies and the spent fuel cask, and for safe disassembly, handling, and reassembly of the reactor vessel head and internals during refueling operations.

The major components of the fuel handling system are the refueling machine, fuel transfer system, spent fuel handling machine, containment polar crane, cask handling crane, new fuel handling crane, and associated fuel and component handling tools.

(1) The refueling machine is a traveling bridge and trolley which spans the refueling pool and is designed to withdraw or insert individual fuel assemblies in the reactor core and transport them to the fuel transfer system. Safe handling of fuel is assured by a system of interlocks on the machine. Thus, the requirements of GDC 61, "Fuel Storage and Handling and Radioactivity Control," and the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Bases," are met.

(2) The fuel transfer system consists of a transfer tube and carriage, two upenders, and the hydraulic power unit. Fuel is transferred between the refueling pool and spent fuel pool through the transfer tube by the carriage. Safe handling of fuel is assured by a system of interlocks in the

system. Thus, the requirements of GDC 61 and the guidelines of Regulatory Guide 1.13 are met.

(3) The spent fuel handling machine is a traveling bridge and trolley which spans the spent fuel pool, refueling canal, and cask storage area and is designed to move individual fuel assemblies between the transfer system and spent fuel storage racks and new fuel elevator, and between the spent fuel storage racks and cask storage area. Safe handling of fuel is assured by a system of interlocks on the machine; thus, the requirements of GDC 61 and the guidelines of Regulatory Guide 1.13 are met.

(4) CESSAR indicates that the containment polar crane, cask handling crane and new fuel handling crane are the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Section 9.1.4 concerning these cranes in order to assure the safety of spent fuel and safe handling of the cask, new fuel, and reactor vessel closure head in accordance with the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification." Additional interfaces are also provided to assure safe fuel handling and thus assure that the CESSAR fuel handling accident analysis is bounding for CESSAR reference plants, thereby protecting against unacceptable radiological releases to the environment (see Section 15.4.6 of this report). These interfaces are in accordance with the requirements of GDC 61 and the guidelines of Regulatory Guide 1.13.

CE has performed a reactor vessel closure head load drop analysis for the CESSAR System 80 plant design. The analysis considered both a flat concentric head drop accident and an offset head drop accident. For both cases, the results indicated that the reactor core is maintained in a coolable configuration, and fuel assembly damage, shutdown cooling supply flow path damage, or vessel support damage would not occur for a drop from a height of 18 feet above the vessel flange. The staff reviewed this analysis during the PDA license review and found it to be conservative and acceptable. Based on that review, we conclude that the CESSAR System 80 vessel head drop analysis complies with the criteria of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and is, therefore, acceptable. In order to assure the validity of the vessel head load drop analysis, CESSAR specifies an interface requirement that referencing applicants assure that the design of the containment polar crane prevents the reactor vessel head from being lifted more than 17 feet above the vessel flange if a single failure could result in dropping of the head. CE further indicates that reactor vessel head handling is the extent of CESSAR involvement with the criteria of NUREG-0612. We will require that reference plant applicants be restricted such that no loads lighter than a single fuel assembly be handled over the open reactor vessel at a height greater than that assumed in the design basis fuel handling accident analysis. Thus, the requirements of GDC 61 are satisfied.

Based on our review of CESSAR, we conclude that the portions of the fuel handling system in the CE scope are in accordance with the requirements of GDC 61 and the guidelines of Regulatory Guide 1.13 and NUREG-0612 with respect to spent fuel handling and handling of the reactor vessel head, and are, therefore, acceptable. We further conclude that the interfaces identified in CESSAR provide adequate information relating to safe fuel handling in order that referencing applicants can comply with the requirements of GDC 2 and 61 and the guidelines of Regulatory Guides 1.13 and 1.29 and NUREG-0612 and are acceptable and complete in this regard.

9.2 Water Systems

9.2.1 Station Service Water System

CESSAR indicates that the design of the station service water system is the responsibility of applicants that reference CESSAR.

9.2.2 Reactor Auxiliaries Cooling Water System

CESSAR indicates that the design of the reactor auxiliaries cooling water system (component cooling water system) is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Sections 5.1.4, 5.4.7, 6A-7, and 9.3.4 relating to provisions for assuring adequate cooling of both essential and nonessential components within the CESSAR scope, including the cooling water requirements for the shutdown cooling heat exchanger in order to assure decay heat removal for safe reactor shutdown and proper safety-related component function during normal and accident conditions. The reactor decay heat loads are based on Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling." These interfaces are in accordance with the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and 44, "Cooling Water," and the guidelines of Regulatory Guides 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components for Nuclear Power Plants," and 1.29, "Seismic Design Classification," and Branch Technical Position ASB 9-2.

CE has submitted Topical Report CENPD 201A, Supplement 1, "System 80 Reactor Coolant Pump Loss of Component Cooling Water Test Report," which summarizes the test method and results for the CESSAR System 80 reactor coolant pump in response to our concern relating to the potential adverse consequences resulting from reactor coolant pump bearing seizure or seal failure in the event of a loss of component cooling water flow to all the pumps as a result of a single failure. The test report indicates that for the CESSAR System 80 reactor coolant pump at least 30 minutes for operator action is available to either restore component cooling water flow or trip the reactor coolant pumps before exceeding design parameters for the bearings or seals. We conclude that the test results are acceptable. The CESSAR FSAR includes interface requirements in Appendix A for the design of the portion of the component cooling water system supplying the reactor coolant pumps including safety-grade indication of loss of component cooling water flow in accordance with our licensing position on this matter.

Based on our review of CESSAR and the interfaces identified, we conclude that CESSAR provides adequate information relating to the reactor auxiliaries cooling water system (component cooling water) in order that referencing applicants can comply with the requirements of GDC 2 and 44 and the guidelines of Regulatory Guides 1.26 and 1.29 and BTP ASB 9-2 and is, therefore, acceptable and complete in this regard.

9.2.3 Demineralized Water Makeup System

CESSAR indicates that the design of the demineralized water makeup system is the responsibility of applicants that reference CESSAR. Primary and secondary water chemistry is discussed in Sections 5.4.2 and 9.3.4 of this report, respectively.

9.2.4 Potable and Sanitary Water Systems

CESSAR indicates that the design of the potable and sanitary water systems is the responsibility of applicants that reference CESSAR.

9.2.5 Ultimate Heat Sink

CESSAR indicates that the design of the ultimate heat sink is the responsibility of applicants that reference CESSAR. However, as indicated in Section 9.2.2 of this SER, CESSAR does include heat load interface requirements for reactor decay heat and components within the CESSAR scope to enable the applicant to design the UHS.

9.2.6 Condensate Storage Facilities

CESSAR indicates that the design of the condensate storage facilities is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Section 5.1.4 for assuring a safety-grade source of water for the emergency feedwater system in order to assure that systems decay heat removal safety function for shutdown during accident and transient conditions is in accordance with the requirements of GDC 2 "Design Bases for Protection Against Natural Phenomena" and 4 "Environmental and Missile Design Bases," and the guidelines of Regulatory Guides 1.26 "Quality Group Classification and Standards for Water-, Steam- and Radioactive Waste Containing Components for Nuclear Power Plants" and 1.29 "Seismic Design Classification."

Based on our review of the CESSAR interfaces, we conclude that CESSAR provides adequate information relating to the condensate storage facilities in order that referencing applicants can comply with the requirements of General Design Criterion 2 and the guidelines of Regulatory Guides 1.26, and 1.29, and is, therefore, acceptable and complete in this regard.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

CESSAR indicates that the design of the compressed air system is the responsibility of applicants that reference CESSAR. In addition, CESSAR identifies compressed air requirements for components within the CE scope. All safety-related systems in the CESSAR scope are designed to perform their intended safety function without the use of instrument air. In addition, CESSAR includes an interface requirement that the air supply to these safety related valves be clean, dry and oil free as further assurance that the CESSAR air operated valves will fail in their proper position upon a loss of air supply. Thus, CESSAR is in accordance with the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.2, concerning protection for safety-related systems against failure in nonsafety-related systems.

Based on our review, we conclude that CESSAR provides adequate information relating to the compressed air system in order that referencing applicants can provide a suitable design in accordance with the requirements of General Design Criterion 2 and the guidelines of Position C.2 of Regulatory Guide 1.29 and is, therefore, acceptable and complete in this regard.

9.3.2 Process Sampling System

The design of the Process Sampling System is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR identifies interface requirements in Sections 5.4.7.1, 6.3.1.3, and 9.3.4.6 relating to process sampling. We have reviewed these interface requirements in the context of the Palo Verde application and found them acceptable. However, we have not, as yet, completed our review with regard to the acceptability and completeness of the interface requirements for future reference plant applications. We will report on the resolution of this issue in a revision to this report.

9.3.3 Equipment and Floor Drainage System

CESSAR System 80 indicates that the design of the equipment and floor drainage system is the responsibility of applicants that reference CESSAR. Protection of safety-related equipment within the CESSAR scope from internal flooding is included in CESSAR interface requirements as discussed in Section 3.4.1 of this SER.

9.3.4 Chemical and Volume Control System

The chemical and volume control system (CVCS) is designed to control and maintain reactor coolant inventory and to control the reactor coolant boron concentration through the process of charging (makeup) and letdown (drawing off). The CVCS purifies the primary coolant by passing letdown flow through heat exchangers and purification ion exchangers. The CVCS is also designed to provide injection flow to the reactor coolant pump seals and to collect the controlled bleedoff from the pump seals. Three positive displacement chargining pumps supply high pressure injection (charging) or borated water into the reactor coolant for normal and emergency boration. The volume control tank serves as a surge volume for the reactor coolant system, to provide for control of hydrogen concentration in the reactor coolant, and to provide a reservoir of makeup for the charging pumps. The boric acid makeup system provides for boron additions to compensate for reactivity changes and to provide shutdown margin for maintenance and refueling operations or emergencies. The boron injection function is required for safe shutdown. The charging portion of the system contains redundant active components and an alternate flow path in order to meet the single failure criterion. The charging and letdown portions of the system are designed to seismic Category I requirements.

The chemical and volume control system (including boron recovery system) includes components and piping associated with the system from the letdown line of the primary system to the charging lines that provide makeup to the primary system and the reactor coolant pump seal water system.

The basis for acceptance in our review has been conformance of the CESSAR design of the CVCS system with the following regulations and regulatory guides: (1) the requirements of GDC 1 and the guidelines of Regulatory Guide 1.26 by assigning quality group classifications to system components in accordance with the importance of the safety function to be performed; (2) the requirements of GDC 2 and the guidelines of Regulatory Guide 1.29 by designing safety-related portions of the system to seismic Category I requirements; (3) the requirements of GDC 14 by maintaining reactor coolant purity and material compatibility to reduce corrosion and thus reduce the probability of abnormal leakage, rapid propagating failure, or gross rupture of the reactor coolant pressure boundary; (4) the requirements of GDC 29 as related to the reliability of the CVCS to provide negative reactivity to the reactor by supplying borated water to the reactor coolant system in the event of anticipated operational occurrences; and (5) the requirements of GDC 60 and 61 with respect to confining radioactivity by venting and collecting drainage from the CVCS components through closed systems.

Based on our review of the proposed system, design bases, and safety classification for the CESSAR chemical and volume control system, and the requirements for system performance of necessary functions during normal, abnormal, and accident conditions, we conclude that the design of the chemical and volume control system and supporting system meet the requirements of GDC 1, 2, 14, 29, 60, and 61 and is, therefore, acceptable.

CESSAR identifies interface requirements for the CVCS with the BOP in Section 9.3.4, which include normal and emergency power; protection from natural phenomena such as floods, winds, tornadoes, and earthquakes; protection from pipe failure and missiles; separation of components; thermal limitations; inspection and testing; materials compatibility; system/component arrangements; radwaste management; overpressure protection; refueling water tank design parameters; alternate source of borated water from the spent fuel pool; fire protection; operating temperature ranges; environmental control; and mechanical interaction between components. We have reviewed these interface requirements in the context of the Palo Verde reference plant application and found them acceptable. However, we have not, as yet, completed our review with regard to the acceptability and completeness of these interface requirements for future reference plant applications. We will report on the resolution of this issue in a revision to this report.

9.4 Heating, Ventilation, and Air Conditioning (HVAC) Systems

CESSAR indicates that the design of heating, ventilation, and air conditioning (HVAC) systems is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Sections 3.11, 5.1.4, 5.4.7, 6B-7, 6A-7, 6.3.1, 7.1.3, and 9.3.4 for assuring a proper operating environment for safety-related systems and components within the CESSAR scope for all modes of operation in accordance with requirements of General Design Criterion 4, "Environmental and Missile Design Bases." Environmental qualification of safety-related systems and components within the CESSAR scope is discussed in Section 3.11 of this SER.

Based on our review of the CESSAR interfaces, we conclude that CESSAR provides adequate information relating to heating, ventilation, and air conditioning systems in order that referencing applicants can comply with the requirements of GDC 4 and is, therefore, acceptable and complete in this regard.

9.5 Other Auxiliary Systems

The design of other auxiliary systems (i.e., fire protection, emergency diesels, communication, and lighting) is the responsibility of applicants that reference CESSAR. However, CESSAR includes interface requirements for each safety-related system which specify that fire protection and emergency power be provided by the reference plant. We will review the fire protection program in each application referencing CESSAR to assure that adequate measures are provided to protect all safety-related systems from fires. The power supply interfaces are discussed in Section 8.1 of this report.

10 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion systems, typically referred to as the "secondary side" of a PWR plant, will be evaluated in the applications that reference CESSAR. These systems consist of the main steam supply system, the turbine-generator, the main condenser, and the condensate and feedwater system. CESSAR identifies interface requirements for these systems, as discussed below, in order to assure that the secondary system will be compatible with the NSSS.

10.2 Main Steam Supply

10.2.1 Main Steam Supply System (Up to and Including the Main Steam Isolation Valves)

CESSAR indicates that the design of the main steam supply system is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Section 5.1.4 relating to the main steam supply system up to and including the main steam isolation valves (including the atmospheric dump and main steam safety valves) in order to assure its proper safety function during decay heat removal, in accordance with the CESSAR accident and transient analyses described in Section 15, under all operating conditions. These interface requirements include assuring main steam isolation during accident and transient conditions and atmospheric dump valve operability to provide a steam relief path for safe shutdown, and are in accordance with the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," 4, "Environmental and Missile Design Bases," 34, "Residual Heat Removal," and the guidelines of Regulatory Guides 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components for Nuclear Power Plants," and 1.29, "Seismic Design Classification."

Based on our review of the CESSAR interfaces, we conclude that CESSAR provides adequate information relating to the main steam supply system in order that referencing applicants can comply with the requirements of GDC 2, 4, and 34 and the guidelines of Regulatory Guides 1.26 and 1.29 and is, therefore, acceptable and complete in this regard.

10.2.2 Turbine Bypass System

The turbine bypass system consists of eight air-operated valves in branch lines downstream of the main steam isolation valves which discharge to the main condenser. This system diverts steam past the turbine, directly to the condenser, in order to control thermal conditions in the primary system when the reactor power is greater than turbine power and prevent the safety valves from opening. These bypass valves have a combined capacity of 55% of full-power steam flow at a steam generator pressure of 1070 psia.

Although this system is not safety-related, it interfaces with certain safety-related aspects of the plant design, via the control systems described in

Section 7.7. Inasmuch as CESSAR only provides a functional description of the turbine bypass system, we will evaluate the system in the context of the overall main steam supply system and power conversion system of the applications that reference CESSAR.

10.3 Circulating Water System

CESSAR indicates that the design of the circulating water system is the responsibility of applicants that reference CESSAR. Protection of safety-related equipment in the CESSAR scope from internal flooding including flooding as a result of failures in the circulating water system is included in CESSAR interface requirements as discussed in Section 3.4 of this report.

10.3.1 Secondary Water Chemistry

The plant Technical Specifications for all pressurized water reactor plants that have been issued an operating license since 1974, either require limiting conditions for operation and have surveillance requirements for secondary water chemistry parameters, or a requirement to establish these provisions after baseline chemistry conditions have been determined. The intent of the provisions was to provide added assurance that the operators of newly licensed plants would properly monitor and control secondary water chemistry to limit corrosion of steam generator components such as tubes and tube support plates.

In a number of instances, the plant Technical Specifications have significantly restricted the operational flexibility of some plants with little or no benefit with regard to limiting degradation of steam generator tube and the tube support plates. Based on this experience and the knowledge gained in recent years, we have concluded that Technical Specification limits are not the most effective way of assuring that steam generator degradation will be minimized.

Due to the complexity of the corrosion phenomena involved and the state-of-the-art as it exists today, we are of the opinion that, in lieu of specifying limiting conditions in the plant Technical Specification, a more effective approach would be to institute a license condition that required the implementation of a secondary water chemistry monitoring and control program containing appropriate procedures and administrative approach controls.

The required program and procedures are to be developed by reference plant applicants with input from CE or other consultants, to account for site and plant-specific factors that affect water chemistry conditions in the steam generators. In our view, plant operation following such procedures would provide assurance that proper attention would be devoted to controlling secondary water chemistry, while also providing the needed flexibility to allow them to deal effectively with an off-normal condition that might arise.

CESSAR provides interface requirements which provide the general provisions for a secondary water chemistry program. This program defines the operational parameters and their limits for the different plant operation modes, and the allowable time span that these chemistry parameters may be out of operational limits. These interface requirements meet our acceptance criteria for non-plant specific details of water chemistry programs, as specified in Branch Technical Position 5-3 which is appended to Standard Review Plan Section 5.4.2.1.

Based on our review of the CESSAR interface requirements, we conclude that the general secondary water chemistry program, including the recommended action for out-of-specification time limits, meet the applicable portions of GDC 14 and are, therefore, acceptable and complete in this regard.

10.4 Condensate and Feedwater System

CESSAR indicates that the design of the condensate and feedwater system is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Section 5.1.4 relating to the main feedwater system in order to assure its proper safety function in accordance with the CESSAR accident analyses. These interface requirements relate to assuring the isolation of main feedwater during accident and transient conditions, as discussed in Section 15, and are in accordance with the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guides 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive Waste-Containing Components for Nuclear Power Plants," and 1.29, "Seismic Design Classification."

CESSAR also includes design provisions and interface requirements in order to prevent the occurrence of unacceptable feedwater/steam generator waterhammer. The CESSAR System 80 steam generator is the preheat type with two separate feedwater injection nozzles on the steam generator, a downcomer nozzle located high on the steam generator used during startup, shutdown, and normal operation, and an economizer nozzle located low on the steam generator used only during normal power operation. The downcomer feedwater nozzle connects to a distribution ring which is provided with top-discharge nozzles in order to preclude rapid draining of the feedwater ring. Further, CESSAR indicates that installation of a downward sloping elbow off both the downcomer and economizer line nozzles at the steam generators will further prevent potential damaging waterhammer by reducing the space available for formation of a relatively large steam pocket within the feedwater piping. An additional interface requires that a check valve be installed in the downcomer feedwater line upstream of the auxiliary feedwater line connection to prevent cold auxiliary feedwater from entering the economizer section of the steam generator and thus precluding potential condensation-induced waterhammer. In response to our concern with testing for waterhammer, CE indicated that interface recommendations for steam generator waterhammer testing could not be incorporated in CESSAR until conclusions on the San Onofre Unit 2 waterhammer incident are available. However, CE has indicated that referencing applicants should as a minimum comply with the waterhammer testing recommendation of NUREG/CR-1606, "An Evaluation of Condensation-Induced Water Hammer in Preheat Steam Generators." Thus, the guidelines of Branch Technical Position ASB 10-2, "Design Guidelines for Water Hammer in Steam Generators with Top Feedring Designs," and the recommendations of NUREG/CR-1606 are satisfied by CESSAR. Any further recommendations concerning waterhammer testing which result from CE's evaluation of the San Onofre Unit 2 incident will be included in CESSAR at a later date.

Based on our review of CESSAR and the CESSAR interfaces, we conclude that CESSAR provides adequate information relating to the condensate and feedwater system in order that referencing applicants can comply with the requirements of GDC 2 and 4 and the guidelines of Regulatory Guides 1.26 and 1.29 and is, therefore, acceptable and complete in this regard.

10.5 Auxiliary (Emergency) Feedwater System

CESSAR indicates that the design of the emergency feedwater system is the responsibility of applicants that reference CESSAR. However, the CESSAR FSAR includes interface requirements in Section 5.1.4 relating to the emergency feedwater system in order to assure its decay heat removal safety function during accident and transient conditions in accordance with the CESSAR accident analyses. These interface requirements describe the functional performance characteristics of the auxiliary feedwater systems, and are in accordance with the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," 4, "Environmental and Missile Design Bases," 19, "Control Room," 44, "Cooling Water," and the guidelines of Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components for Nuclear Power Plants," 1.29, "Seismic Design Classification," and 1.62, "Manual Initiation of Protective Actions," and Branch Technical Position ASB 10-1.

Based on our review of the CESSAR interfaces, we conclude that CESSAR provides adequate information relating to the emergency feedwater system in order that referencing applicants can comply with the requirements of GDC 2, 4, 19, and 44 and the guidelines of Regulatory Guides 1.26, 1.29, and 1.62 and Branch Technical Position ASB 10-1 and is, therefore, acceptable and complete in this regard.

11 RADIOACTIVE WASTE MANAGEMENT

11.1 Source Terms

Radioactive materials may be released to the environment from the gaseous waste processing systems, liquid waste processing system, the boron recycle system (BRS), the steam generator blowdown system, and the turbine building floor drain system at a nuclear plant utilizing a pressurized water reactor. Of these, only the BRS as a part of the Chemical and Volume Control System (CVCS) is within the standard scope of the CESSAR design.

CESSAR does not include the radioactive waste management systems in its design scope. These systems will be provided by the reference plant. However, the CESSAR does include, as interface information, the concentrations of radioactive materials in the primary coolant, the applicable bases for evaluating radionuclide concentrations in the secondary system and the flow rates of streams that are input to the radwaste management systems. This interface information will be used (1) as a design basis for coolant source terms for evaluating gaseous and liquid effluent releases in CESSAR reference plants during normal operation and anticipated operational occurrences, and (2) as a design basis for evaluating liquid effluent releases from the BRS during normal operation and anticipated operational occurrences, as appropriate.

The principal parameters used for calculating primary and secondary coolant concentrations are given in Table 11-1; the principal parameters for the BRS are given in Table 11-2. Detailed description of the BRS is given in Section 9.3.4.2 and Tables 9.3-3 and 9.3-4 of the CESSAR FSAR.

11.2 System Description and Evaluation of the BRS

The BRS processes the shim bleed which has already passed through the letdown purification filter and the letdown purification mixed bed demineralizer, along with the reactor-grade water collected in the equipment and reactor drain tanks. These streams are processed by a pre-holdup mixed bed demineralizer, a gas stripper, a boric acid concentrator (an evaporator) and a boric acid condensate anion demineralizer. The processed liquid is used as makeup water water in the plant. Although the system is designed to recycle all the processed liquid, small fraction of the processed liquid may be discharged due to operational upsets or for controlling the tritium inventory in the plant. Spent demineralizer resins, evaporator concentrates and the filters from the BRS will be periodically transferred to the solid waste management for eventual shipment off-site to a licensed burial facility.

The BRS, which is operated in the batch mode, has a capacity to process approximately 29,000 gallons/day. Based on the information provided in the FSAR for CESSAR System 80, we assume that the shim bleed input to the BRS averaged on a yearly basis during plant shutdowns, startups and boron dilution over core life is approximately 3590 gallons/day, which includes a daily average of 720 gallons/day during boron dilution over core life; reactor and equipment drain tanks input to the BRS is approximately 250 gallons/day. The holdup tank

TABLE 11-1 Principal Parameters Used for Calculating Expected Concentrations of Radionuclides in Primary and Secondary Coolants for CESSAR System 80 Final Design

Reactor power level (megawatts thermal)	3800
Failed fuel percentage	0.12
Mass of primary coolant (pounds)	5.71×10^5
Primary coolant purification letdown rate (gallons/minute)	72
Shim bleed rate (gallons/minute)	0.5
Leak rate to secondary system (pounds/day)	100
Steam flow rate (pounds per hour)	1.72×10^7
Number of steam generators (U-Tube)	2
Mass of liquid per steam generator (pounds)	1.67×10^5
Decontamination factors for the letdown demineralizer	
Iodine	10
Cesium and Rubidium	2
Tritium and noble gases	1
Others	10
Steam generator carry over factor	
Noble gases and tritium	1.0
Bromines and iodines	0.01
All others	0.001

[1]No continuous gas stripping of the full letdown flow is assumed; however, the streams processed by the boron recycle system are stripped of gases.

[2]Credit is taken for radionuclide removal by an additional letdown purification demineralizer which is intermittently used (20 percent of the time).

TABLE 11-2 Parameters of Principal Components of the Boron Recycle System

Component	Number	Capacity	
		Expected	Design
Pre-holdup demineralizer	1	72 gallons/minute	125 gallons/minute
Boric acid concentrator	1	20 gallons/minute	20 gallons/minute
Boric acid condensate demineralizer	1	20 gallons/minute	100 gallons/minute
Gas stripper	1		140 gallons/minute
Holdup tank	1		435,000 gallons (minimum)

Decontamination Factors for the Boron Recycle System

Radionuclide	Shim Bleed Input*	Equipment and Reactor Drain Tank Input
Iodine, Bromine	10^6	10^5
Cesium and rubidium	4×10^3	2×10^3
Tritium	1	1
Others	10^5	10^4

*This includes the decontamination factor due to letdown purification mixed bed demineralizer. It does not include additional credit for radionuclide removal by a second letdown purification demineralizer since it may be used only intermittently (20 percent of the time); however, it is factored in the evaluation of primary coolant concentrations.

storage capacity (435,000 gallons) and the difference between the expected daily input rate and the system design capacity provide adequate margin for processing surge flows.

Based on the above considerations, we conclude that the design and capacity of the BRS is acceptable.

11.3 Conformance with NRC Regulations and Staff Positions

We have reviewed the BRS and concluded that the system design and capacity are adequate to control the release of radioactive materials in liquid effluents from the BRS during normal operation and anticipated operational occurrences in accordance with GDC 60, and is, therefore, acceptable. Further, we have reviewed the interface information applicable to the waste management systems and conclude that it is consistent with Regulatory Guides 1.70 and 1.112 and the applicable criteria given in the Standard Review Plan, Section 11.1, "Source Terms," and is, therefore, acceptable and complete in this regard. This information will be used in our evaluation of the radwaste management systems in applications referencing CESSAR.

12 RADIATION PROTECTION

We have evaluated the proposed radiation protection program presented in the CESSAR FSAR. The radiation protection measures within the CESSAR scope are intended to ensure that internal and external occupational radiation exposures and exposure of the population due to station conditions, including anticipated operational occurrences, will be as low as is reasonably achievable and within the limits of 10 CFR Part 20. In the CESSAR FSAR Chapter 12, CE discusses design features, provides basic radiation sources for shielding design of equipment and components, describes equipment and system design features for control of onsite exposures, such as reduction of crud buildup and facility decontamination features, and describes the administrative controls to be employed throughout all phases of the plant design to ensure that the intent of Regulatory Guide 8.8 is met and that personnel radiation exposures will be maintained as low as is reasonably achievable (ALARA). Radiation protection design reviews for reference plant applications will take a place prior to the release of design drawings, system or components design requirements, and follow-up reviews will be conducted to ensure resolution within the established radiation protection guidelines and to maintain personnel radiation exposures ALARA.

The basis of our acceptance of the CESSAR material is that doses to personnel will be maintained within the limits of 10 CFR Part 20, "Standard for Protection Against Radiation." The reference plant radiation protection design and program features must also be consistent with the guidelines of Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable" (Revision 3). In response to our review questions, CE has amended applicable sections of Chapter 12 of the FSAR and has added design features for assuring that occupational radiation exposures are as low as reasonably achievable. The use of design techniques and features to minimize radiation exposure from activated corrosion products and the use of separation of radioactive components, remotely operated valves, remotely replaceable filter cartridges, hydraulically removable spent resin from ion exchangers, skid mounted equipement for quick removal to low radiation area for maintenance or repair, are examples of such person-rem reduction features.

On the basis of our review of the CESSAR FSAR we conclude that implementation of the radiation protection measures incorporated in the CESSAR design will provide reasonable assurance that personnel doses are maintained as low as is reasonably achievable. Design is such that the personnel doses should be below the limits established by 10 CFR Part 20 and the design features are consistent with the guidelines of Regulatory Guide 8.8 (Revision 3).

12.1 Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)

This subsection describes the radiation protection measures incorporated in the CESSAR design to ensure that internal and external radiation exposures to station personnel, contractors and the general population due to station conditions, including anticipated operational occurrences, will be within all applicable limits, and furthermore, will be as low as is reasonably achievable (ALARA).

CESSAR incorporates the following facility and equipment major design considerations in order to satisfy the above listed radiation protection design objectives.

- Experience from past designs and operating reactors has been employed in evaluating the performance of plant systems in mitigating radioactive buildup, and in reducing radiation levels.
- Systems and equipment employed in CESSAR have been designed with the objective of reducing the need for maintenance within radiation areas.
- Whenever possible materials were selected to withstand a 40-year service life thus minimizing the need for replacement and reducing maintenance frequencies.
- Controls are located in low radiation zones.
- Equipment such as heat exchangers and valves are designed for ease of access during maintenance. Equipment is environmentally qualified to meet their performance requirements under the environmental and operating conditions in which they will be required to function.
- Significant reduction of personnel exposure during inservice inspection has been accomplished by the reduction in weld footage. This has been accomplished by component redesign, use of forged sections versus forged-welded plate sections, and increasing the size of certain sections.
- The design incorporates ALARA considerations for plant decommissioning.

The CESSAR design considerations conform with the guidelines of Regulatory Guide 8.8 and are, therefore, acceptable.

12.2 Radiation Sources

This subsection discusses and identifies the sources of radiation that form the basis for shield design calculations for the design of personnel protection measures and for dose assessments.

1. Containment Sources

   The shielding design source terms were based on full-power operation with 1% fuel cladding defects. Sources in the primary coolant include fission products released from fuel clad defects and activation and corrosion products. During plant operation Nitrogen-16 has been identified to be the primary radiation source for shielding design throughout most of the reactor coolant system.

   Maximum neutron and gamma spectra outside the reactor vessel during operation and shutdown, and the N-16 activity at various locations in the primary loop are also provided. The spent fuel gamma source is provided and the isotopic composition of design sources is also provided for the reactor coolant system for the expected long-lived crud activity, spent fuel pool water, and for various systems and equipment such as CVCS heat exchangers, ion exchangers filters, tanks, and shut down cooling system.

The source terms used for plant shielding design were based on 1% fuel cladding defect, which is a factor of 4 higher than that required by the Standard Review Plan, and are acceptable.

2. Airborne Radioactive Material Sources

This subject is not within the CESSAR scope and will be evaluated in applications referencing CESSAR.

12.3 Radiation Protection Design Features

The radiation protection design features described in CESSAR are intended to help maintain occupational radiation exposures as low as is reasonably achievable. Many of these design features have been incorporated as a result of the CE radiation protection design review and from radiation exposure experience gained during the design of other nuclear power plants. Following are some of the examples which will help to reduce radiation exposures to personnel:

- Pumps
  - Most pumps and associated piping are flanged to facilitate ease of removal to a low radiation area for maintenance.
  - All pump casings are provided with drain connections to facilitate decontamination.
- Valves
  - Radiation resistant seals, gaskets and elastomers are employed when practical to extend the design life and reduce maintenance requirements
  - Power operated valves in the primary system are provided with double packing, a lantern gland and stem leakoffs to collect leakage and to direct radioactive fluid away from access areas. All valve packing glands have provisions to adjust packing commpression to reduce leakage.
  - Valves are designed so that they may be repaced without removing the yoke or topworks.
  - Remotely operated values are utilized where practical and necessary.
  - Valve wetted parts are made of austenitic stainless steel or other corrosion resistant material.
  - Low leakage valves with backseats are employed wherever possible. Packless diaphragm valves are employed in highly contaminated systems.

- Liquid filters

    - Filter housings are provided with vent connections and designed for complete drainage.

    - Filter housings are designed with a minimum of crevices in order not to accumulate radioactive crud.

    - Filter housings and cartridges are designed to permit remote removal of the filter elements.

- Tanks

    - Tanks are designed to be isolated for maintenance and provisions will be made for complete drainage.

    - Tanks are provided with at least one of the following means of cleaning the tank internals for decontamination purposes:

        a. Ample space is provided to facilitate cleaning from the tank manway.

        b. Internal spray nozzles are provided on potentially highly contaminated tanks for internal decontamination.

        c. The ability to back flush or drain inlet screens hydraulically will be provided (on tanks or vessels with these screens) to facilitate decontamination.

    - All tanks are vented to either the gas collection header or the gas surge header which will facilitate removal of potentially radioactive gases during maintenance.

    - Non-pressurized tanks are provided with overflows, routed to a floor drain or other suitable collection point to avoid radioactive fluids spilling to the floor or ground.

    - Tanks are designed with a minimum of crevices in order not to accumulate radioactive crud.

- Ion exchangers

    - Ion exchangers are designed for complete drainage.

    - Spent resin removal is designed to be done remotely by hydraulically flushing the resin from the vessel to the Solid Waste Management System.

    - The fresh resin inlet is designed to extend into a low radiation area above the shielded compartment housing the ion exchanger.

    - Ion exchangers are designed with a minimum of crevices in order not to accumulate radioactive crud.

- Heat exchangers

  - Heat exchangers are designed to accommodate the requirements of in-service inspection and for ease of access during maintenance to reduce the time operators are required to spend in a radiation environment.

  - Materials are selected to minimize the need for replacement and to reduce maintenance frequencies; corrosion resistant materials are employed.

- Package Units

  - Each package unit is skid mounted with all motors and pumps located on the periphery of the skid for free access and for quick removal to a low radiation areas for maintenance or repair.

  - Space is provided on the skid for placement of portable shielding.

  - All package components are provided with provisions for flushing, drawing, and chemical cleaning.

  - Heat exchangers are readily accessible for maintenance.

  - Controls are remotely mounted and the package will be able to be remotely monitored. As many control elements as possible are mounted remotely from the components.

  - Components are designed with a minimum of crevices in order not to accumulate radioactive crud.

  - Radioactive gas is collected and sent to the Gaseous Waste Management System.

- Reactor Vessel Head Vent

  - A vent nozzle and line is provided on the reactor vessel head. Utilization of this design feature will allow a reduction of exposure during the head removal process by minimizing the gases discharged directly to the containment atmosphere while the head is being removed.

- Reactor Coolant System Leakage Control

  - Exposures from airborne radionuclides to personnel entering the containment will be minimized by controlling the amount of reactor coolant leakage to the containment atmosphere. Examples of a controlled leakage are listed below:

    a. Primary pressurizer safety valve leakage is directed to the Reactor Drain Tank.

b. Valves larger than 2" in diameter are provided with a double-packed stem with an intermediate lantern ring with a leak-off connection to the Reactor Drain Tank.

c. Instrumentation is provided to detect abnormal reactor coolant pump seal leakage. The reactor coolant pumps are equipped with two stages of seals plus a vapor or backup seal. The vapor or backup seal will prevent leakage to the containment atmosphere and allow sufficient pressure to be maintained to direct the controlled seal leakage to the Volume Control and Reactor Drain Tanks. The vapor seal is designed to withstand full Reactor Coolant System pressure in the event of failure of any or all of the two primary seals.

° Refueling Equipment

- All spent fuel transfer and storage operations are designed to be conducted underwater to insure adequate shielding and to limit the maximum continuous radiation levels in working areas.

- The equipment is designed to prevent the fuel from being lifted above the minimum safe water depth, thereby limiting personnel exposures and avoiding fuel damage.

- The equipment design limits the possibility of inadvertent fuel drops which could cause fuel damage and personnel exposures.

- The refueling equipment design will facilitate the transfer of new and spent fuel at the same time to reduce overall fuel handling time; and, therefore, personnel exposures during refueling.

- Underwater cameras are used to facilitate safe handling and visual control, thus minimizing errors and potential exposures.

- Equipment is provided to allow for the under water determination of leaking fuel elements.

° Remote Instrumentation

- All systems containing radioactive fluids are designed to be controlled remotely to the maximum extent practical. This will allow personnel radiation exposures from the normal operation of these systems to be minimized.

° Inservice Inspection Equipment

- Inspection of the reactor coolant pressure boundary can be done with remote equipment to keep personnel exposures to a minimum.

° Inservice Inspection of Reactor Vessel Nozzle Welds

- The design of welds joining the reactor vessel nozzle to reactor coolant pipe permits inservice inspection to be accomplished from the inside of the reactor vessel. Automated equipment normally used for reactor vessel pressure boundary inspections can be utilized in this area.

- In the event inservice inspection of this area is performed from the outside, insulation for the reactor vessel and reactor coolant piping utilizes removable sections for access. These removable sections are lightweight and are held in place mainly by quick actuation type buckle fasteners. After the necessary panels are removed, remote equipment can be utilized to perform the required inspections.

o Material Selection

Material is selected as described below to reduce exposures by reducing maintenance frequencies and by providing less circulating crud as a source of exposure where maintenance will be necessary.

a. Materials of construction for components containing radioactive materials will be selected with consideration of potential release of activiated corrosion products form these materials.

b. Radiation exposure levels were considered when selecting materials for 40-year service.

c. Material selection was made with consideration given to other fluid conditions which could lead to premature material failure.

Additional examples of CESSAR efforts to maintain radiation exposures to personnel ALARA are given in subsection 12.1 of this report.

These design features are consistent with those in Regulatory Guide 8.8 and are, therefore, acceptable.

Based on the information presented in CESSAR, we conclude that CE has described radiation protection design features that are consistant with maintaining in-plant radiation exposures within the applicable limits of 10 CFR Part 20 and maintaining exposures as low as is reasonably achievable.

13 CONDUCT OF OPERATIONS

The utility organization, training, procedures, physical security, and emergency planning will be evaluated in the applications that reference CESSAR.

14 INITIAL TEST PROGRAM

CESSAR does not describe a complete initial test program. A description of the initial test program for the balance-of-plant must be provided in the applications referencing CESSAR.

The objectives of the CESSAR scope of the initial test program are to:

1. Demonstrate that components and systems of the NSSS operate in accordance with design requirements.

2. Demonstrate that the NSSS can be safely operated and that performance levels can be maintained in accordance with established safety requirements.

3. Confirm proper transient system operation and thereby verify that the NSSS can be brought to power as well as to a shutdown condition in a controlled and safe manner.

4. Provide verification of core physics parameters and baseline performance data for use during normal plant operation.

The initial test program begins as systems become available for testing during the construction phase and ends with compl+ n of the power ascension tests. The testing program is divided into the following major tests: Preoperational Tests; Integrated Reactor Coolant System (RCS) Heatup and Pre-Fuel Loading (Precore) Hot Functional Tests; Initial Core Loading; Post-Fuel Loading (Post Core) Hot Functional Tests; Initial Criticality; Low Power Physics Tests; and Power Ascension Tests.

Preoperational tests, listed in Table 14-1, are performed to demonstrate proper system and component installation, calibration and operation; to demonstrate the capability of systems and components to meet safety-related performance requirements; and to provide initial baseline performance data for use during subsequent plant operation. Simulated signals or inputs may be used where actual process signals are not available to verify system and instrument operating ranges.

The Integrated RCS Heatup and Pre-Fuel Loading Hot Functional Tests, listed in Table 14.2, are performed to assure, where possible, that systems necessary for normal plant operation will safely perform their function when required. During these Hot Functional Tests, plant operating procedures are used to the extent possible to bring the plant to normal operating temperature and pressure. Upon completion of these tests, plant operating procedures are used, as practical, to bring the plant to cold shutdown conditions. This testing sequence provides system baseline performance data and operator familiarity with the plant operating procedures. When this phase of testing is completed and results reviewed and approved, the plant is ready for fuel loading.

The initial fuel loading phase of the startup test program provides a step-by-step process for safely accomplishing and verifying fuel loading. Temporary instrumentation will be installed during fuel-loading to supplement the permanently installed count-rate instrumentation.

The Post-Fuel Loading Hot Functional Tests, listed in Table 14-3, are performed prior to initial criticality. The objectives of these tests are to provide additional assurances that plant systems necessary for normal plant operation function as expected, and to obtain performance data on core-related systems and components such as control rods. Normal plant heatup procedures are used to the extent practical to bring the plant from cold shutdown conditions to normal operating temperature and pressure.

The Initial Criticality phase of the startup test program assures that criticality is achieved in a safe and controlled manner. Core reactivity conditions are monitored continuously during the approach to criticality to assure that the core responds as predicted and to assure a cautious approach to criticality.

Following Initial Criticality, a series of Low Power Physics Tests, listed in Table 14-4, are performed to verify selected core design parameters. These tests serve to substantiate conservatism in Safety Analyses and Technical Specifications. They also demonstrate that core characteristics are within expected limits and provide data for benchmarking the computer algorithms used for predicted core characteristics later in life. A series of Power Ascension Tests, listed in Table 14-5, is conducted to bring the reactor to full power. Testing is is performed at plateaus of approximately 20, 50, 80, and 100 percent rated power to demonstrate that the facility operates in accordance with its design during steady-state conditions and to the extent practicable during operational transients. For the "first-of-a-kind" plant, the Power Ascension Tests will be expanded to validate the design methods and to verify new design concepts.

Our review of Chapter 14 of the CESSAR FSAR was conducted in accordance with the Standard Review Plan. We reexamined the Safety Evaluation Report that was issued at the completion of our Preliminary Design Review to determine the principal design criteria for the plant and to identify any specific concerns or unique features that would warrant special testing. Chapters 1 through 12 of CESSAR FSAR were reviewed for familiarization with the facility design and nomenclature. Chapter 15 was reviewed to identify assumptions pertaining to performance characteristics that should be verified by testing, and to identify all structures, systems, and components and design features that were assumed to function (either explicitly or implicitly) in the accident analysis. Licensee Event Reports for operating reactors of similar design were reviewed to identify potentially serious events and chronic or generic problems that might warrant special test consideration. Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors (NUREG-0212) were reviewed to identify structures, systems, and components that would be relied upon for establishing conformance with safety limits or limiting conditions for operations. And finally, the Startup Test Reports for other pressurized water reactor plants were reviewed to identify problem areas that should be emphasized in the CESSAR FSAR initial test program.

Our review of the initial test program confirmed that:

1. CE will provide reference plant applicants with guidelines for preparation of detailed test procedures.

2. CE will maintain an on-going effort to continually feedback to its startup organization operating and test experiences from other facilities.

3. Test descriptions are provided and include all structures systems, components, and design features within the scope of CESSAR that:

   (a) will be used for shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period;

   (b) will be used for shutdown and cooldown of the reactor under transient (infrequent or moderate-frequency events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions;

   (c) will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility Technical Specifications;

   (d) are classified as engineered safety features or will be relied upon to ensure the operability of engineered safety features within design limits;

   (e) are assumed to function or for which credit is taken in the accident analysis; and

   (f) will be used to process, store, control, or limit the release of radioactive materials.

4. The test objectives, prerequisites, test methods, and acceptance criteria for each test description are in sufficient detail to establish that the test will verify adequacy of the structures, systems, and components.

5. The test program conforms with applicable Regulatory Guides, or that adequate justification was provided for all exceptions.

   These Regulatory Guides include: Regulatory Guides 1.20 (December 1971), "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing"; 1.68 (November 1973), "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors"; 1.68.2 (July 1978), "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants"; and 1.70 (June 1974), "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors".

CE made a number of changes to the initial test program as a result of our review. Examples of these changes follow:

1. Acceptance criteria for the major preoperational and startup tests were expanded to ensure that quantifiable and referenceable criteria are available to establish that the actual test objectives are met.

2. The Initial Criticality description was modified to provide more rigorous requirements on expected boron concentration predictions, CEA position calculations, and startup rate limits. Testing requirements were added to demonstrate adequate overlap of source-and intermediate-range neutron instrumentation.

3. Tests were added to include 125% of rated static loads and 150% of rated dynamic loads for the refueling hoist, reactor vessel heat lifting rig, reactor internals lifting rig, and other hoists or lifting equipment.

4. The Low Pressure Safety Injection subsystem test was modified to include a full-scale model test of the recirculation mode of operation to verify vortex control.

5. Testing was added to more accurately determine the Reactor Protection System response times.

6. Tests were added to perform at least three additional measurements of control element assembly drop times for all CEA's outside the two-sigma limit.

7. Testing was added, for the first-of-a-kind CESSAR plant to demonstrate adequate shutdown margin with the CEA of greatest reactivity worth stuck out of the core.

8. The Loss-of-Offsite-Power Test was modified to specify that the power loss be maintained for at least thirty minutes after reactor shutdown.

9. The test of remote reactor plant shutdown capability was modified to include only the equipment for which credit is taken to perform a remote shutdown, and to ensure that the reactor trip initiation be performed outside the control room.

Based on our review we have concluded that the initial test program described in the CESSAR FSAR complies with the acceptance criteria of Section 14.2 of the Standard Review Plan and that successful completion of the program will demonstrate the functional adequacy of plant structures, systems, and components within the scope of CESSAR. We also have concluded that the requirements of GDC 1 and Section IV of 10 CFR 50 Appendix B will be met for that portion of a facility's initial test program which is described in CESSAR.

This review and evaluation was performed with the assistance of Battelle Pacific Northwest Laboratories personnel.

Table 14-1

CESSAR
PREOPERATIONAL TESTS

Reactor Coolant Pump Motor Initial Operation
Reactor Coolant System Test
Pressurizer Safety Valve Test
Pressurizer Pressure and Level Control System Test
CVCS Letdown Subsystem Test
CVCS Purification Subsystem Test
Volume Control Tank Subsystem Test
CVCS Charging Subsystem Test
Chemical Addition Subsystem Test
Reactor Drain Tank Subsystem Test
Equipment Drain Tank Subsystem Test
Boric Acid Batching Tank Subsystem Test
Gas Stripper Subsystem Test
Boronmeter Subsystem Test
Letdown Process Radiation Monitor Test
Gas Stripper Radiation Monitor Test
Shutdown Cooling Subsystem Test
High Pressure Safety Injection Subsystem Test
Low Pressure Safety Injection Subsystem Test
Safety Injection Tank Subsystem Test
Engineered Safety Features Auxiliary
Relay Cabinet Test
Plant Protection System Test
Excore Nuclear Instrumentation System Test
Fixed Incore Nuclear Signal Channel Test
Moveable Incore Detector Drive System Test
Control Element Drive Mechanism Control System Test
Reactor Regulating System Test

Concentrated Boric Acid Subsystem Test Reactor Makeup Subsystem Test Holdup Subsystem
Test Boric Acid Concentrator Subsystem Test
Steam Bypass Control System Test
Feedwater Control System Test
Core Operating Limit Supervisory System Test
Reactor Power Cutback System Test
Refueling Equipment Test

Table 14-2

CESSAR
PRE-FUEL LOAD HOT FUNCTIONAL TESTS

Precore Hot Functional Test Controlling Document (Integrated Systems Operation Tests)
Test Data Record (Instrumentation Cross-Checks)
Reactor Coolant System Expansion Measurements
Reactor Coolant and Secondary Water Chemistry Data
Pressurizer Performance
Control Element Drive Mechanism Performance
Instrument Correlation
Reactor Coolant System Flow Measurements
Reactor Coolant System Heat Loss
Reactor Coolant System Leak Rate Measurement
Chemical and Volume Control System Integrated Test
Safety Injection Check Valve Test
Boration/Dilution Measurements
Postcore Hot Functional Test Controlling Document Tests (Integrated Primary, Secondary, and Auxiliary Systems Operational Tests)
Instrument Correlation
Reactor Coolant System Flow Measurements
Control Element Drive Mechanism Performance
Reactor Coolant and Secondary Water Chemistry Data
Pressurizer Spray Valve and Control Adjustments
Reactor Coolant System Leak Rate Measurement
Incore Instrumentation

Table 14-3

CESSAR
POST-FUEL LOAD HOT FUNCTIONAL TESTS

Postcore Hot Functional Test Controlling
Document Tests (Integrated Primary, Secondary, and Auxiliary Systems
Operational Tests)
Instrument Correlation
Reactor Coolant System Flow Measurements
Control Element Drive Mechanism Performance
Reactor Coolant and Secondary Water Chemistry Data
Pressurizer Spray Valve and Control Adjustments
Reactor Coolant System Leak Rate Measurement
Incore Instrumentation

TABLE 14-4

CESSAR
LOW POWER PHYSICS TESTS

Test Title	First-of-a-kind*	Follow-On Units***
Low Power Biological Shield Survey Test	320°F/565°F	565°F
CEA Coupling/Symmetry Test**	320°F/565°F	565°F
Isothermal Temperature Coefficient Test	320°F-565°F	565°F
Regulating CEA Group Worth Test	320°F & 565°F	565°F
Shutdown CEA Group Worth Test	565°F	565°F
Differential Boron Worth Test	320°F & 565°F	565°F
Critical Boron Concentration Test	320°F-565°F	565°F
Pseudo Dropped and Ejected CEA Worth Test	565°F	N/A

*An expanded test program is conducted for the "first-of-a-kind" to validate the design, the design methods, and the safety analysis assumptions.

**On the "first-of-a-kind" plant the CEA coupling check is performed at 320°F and the CEA symmetry test is performed at 565°F.

***Reduced testing is contingent upon the demonstration that "Follow-On" plants behave in an identical manner as the first of a kind plant.

TABLE 14-5

CESSAR
POWER ASCENSION TEST

Test Title	First-of-a-Kind*	Follow-On Units**
Natural Circulation Test	***$\geq$80%	N/A
Variable T_{avg} (Isothermal Temperature Coefficient & Power Coefficient) Test	20,50,80,100%	50% & 100%
Unit Load Transient Test	Post 100%	Post 100%
Control Systems Checkout Test	50,80%	50,80%
RCS and Secondary Chemistry and Radiochemistry Test	20,50,80,100%	20,50,80,100%
Turbine Trip Test	100%	100%
Unit Load Rejection Test	Post 100%	Post 100%
Shutdown from Outside the Control Room Test	$\geq$10%	$\geq$10%
Loss of Offsite Power Test	$\geq$10%	$\geq$10%
Biological Shield Survey Test	20,50,80,100%	20,50,80,100%
Xenon Oscillation Control Test	65%	N/A
Dropped CEA Test	Post 100%	N/A
"Ejected" CEA Test	Post 100%	N/A
Steady-State Core Performance Test	20,50,80,100%	20,50,80,100%
Intercomparison of PPS, CPC and Process Computer Inputs	20,50,80,100%	20,50,80,100%
Verification of CPC Power Distribution Related Constants	20,50,80,100% and post 100%	20,50,80,100% and post 100%

TABLE 14-5 (Continued)

Test Title	First-of-a-Kind*	Follow-On Units**
Main and Emergency Feedwater	***$\geq$10%	$\geq$10%
CPC Verification	20,50,80,100%	20,50,80,100%
Steam Dump and Bypass Valve Capacity Test	$\geq$15%	$\geq$15%
Incore Detector Test	20,50,80,100%	20,50,80,100%
COLSS Verification	20,50,80,100%	20,50,80,100%

*An expanded test program is conducted for the "first-of-a-kind" to validate the design, the design methods, and the safety analysis assumptions.

**Reduced testing is contingent upon the demonstration that "Follow-On" plants behave in an identical manner as the "first-of-a-kind" plant.

***Initial Power Level

15 ACCIDENT AND TRANSIENT ANALYSIS

15.1 Introduction and Analytical Techniques

We have evaluated the response of CESSAR to postulated disturbances in process variables and to postulated malfuctions or failures of equipment. The potential consequences of each event are examined to determine their effect on the plant, to determine whether plant protection systems are adequate to limit consequences of such occurrences, and to ensure that the design criteria of the applicable regulations are met. The criteria set forth in NUREG-0800 (Standard Review Plan) are considered an acceptable means for meeting the regulations.

Initial plant conditions for the safety analyses are given in Table 15.1-1. This range of initial conditions corresponds to a range compatible with the monitoring functions of the core operating limit supervisory system (COLSS) which is a nonsafety-related instrumentation system that aids the operator in maintaining the plant within the limiting conditions of operation (LCO). COLSS monitoring and calculational functions include peak linear heat rate, margin to departure from nucleate boiling (DNB), total core power, and azimuthal tilt. COLSS compares these parameters to their LCOs and provides an alarm to the operator via the plant computer if an LCO is approached or exceeded, as discussed in Section 7.7.10 of this report.

A range of fuel parameters based on first-core values are used for the safety analyses. These include Doppler weighting factors from 0.85 to 1.15, moderator temperature coefficients from 0.0 $\delta\rho/^{o}F$ to 3.5×10^{-4} $\delta\rho/^{o}F$, shutdown control element assembly (CEA) reactivity worth available at full power and zero power at 10.0% $\delta\rho$ and 6.4% $\delta\rho$, respectively, and decay heat generation rate based upon an infinite reactor operating period at full power. The decay heat curve used in the analyses is that required by 10 CFR 50, Appendix K. The reactivity insertion curve, used to represent the control assembly insertion, accounts for a stuck rod, in accordance with GDC 27.

CE-1 is the DNB correlation used to determine thermal margins in the transient analyses. The applicability of CE-1 is discussed in Section 4.4 of this report.

The reactor protection system (RPS) trips considered in the analyses in accordance with GDC 20 are:

(1) High logarithmic power level

(2) Variable overpower

(3) High pressurizer pressure

(4) Low pressurizer pressure

(5) Low steam generator pressure

(6) Low steam generator water level

Table 15.1-1 Initial Condition Range Considered in the Safety Analyses

Parameter	Units	Range
Core power	% of 3800 Mwt	0-102
Radial one-pin peaking factor (with uncertainty)	-	to 1.63
Axial shape index[1]		$-0.3 \leq ASI \leq +0.3$
Core inlet coolant flowrate	% of 157.4×10^6 lbm/hr	95-116
Pressurizer water level	% distance between upper tap and lower tap above lower tap	26 to 60
Core inlet coolant[2] temperature	F	500-580
Reactor coolant system pressure	psia	1785-2400
Steam generator water level	% distance between upper tap and lower tap above lower tap	40-88

$$^{1}ASI = \frac{\text{area under axial shape in lower half of core} - \text{area under axial shape in upper half of core}}{\text{total area under axial shape}}$$

[2]Additional restrictions were applied to Sections 15.2.3 and 15.1.5; minimum core inlet coolant temperature equals 560°F; maximum core inlet coolant temperature equals 570°F.

(7) High steam generator water level

(8) Low departure from nucleate boiling ratio (DNBR)

(9) High local power density

Time delays to trip are included in the analyses. The CPC system consists of four digital calculators (one in each RPS protection channel) which calculate DNBR and local power density. These values are compared with trip setpoints for initiation of a low DNBR trip and high local power density trip, as described in Section 7.2.1.3 of this report.

The low DNBR trip is provided to trip the reactor core when the calculated DNBR approaches a preset value. The algorithms which calculate the minimum DNBR include allowance for sensor and processing time delays and uncertainties. Many events as analyzed in Chapter 15 of the FSAR have their minimum DNBR reach exactly 1.19 as calculated by the CE-1 correlation. Details of the Reactor Protection System and Engineered Safety Features Actuation System are addressed in Sections 7.2 and 7.3 of this report. The staff evaluation of the CPC system are addressed in Section 4.4 and Section 7.2 of this report.

The analysis methods used for postulated transients and accidents are normally reviewed on a generic basis. In this regard, we have received submittals from CE of the computer codes and methods used in the analysis of reactor transients as shown in Table 15.1-2. The mathematical model used in the steam line break and feedwater line break analyses is described in the CESSAR FSAR, as discussed below. The CE topical reports associated with the thermal-hydraulic design of the CESSAR reactor core are discussed in Section 4.4 of this report.

Generic topical reports of methods for analysis of steam and feed line breaks have been submitted for staff approval by CE in appendices to the CESSAR FSAR. Information specific to CESSAR steam and feedline break analysis has been submitted by CE. Our review of this information is not yet complete. However, the results of our review to date indicate that there is reasonable assurance that the conclusions drawn on these analyses should not be appreciably altered by completion of the analytical review. If the final approval of the analytical methods indicates that any revisions to the analyses are required, CE will be required to implement the results of such changes in CESSAR.

The topical reports on the methods used in the analysis of reactor transients are under review by the staff. The status of these code reviews is listed below:

(1) CENPD-107 CESEC - Digital Simulation of A Combustion Engineering Nuclear Steam Supply System, April 1974

The CESEC computer program is used for the analysis of various system transients and is currently under review by the staff. If final approval of CENPD-107 indicates that any revisions to the analyses are required, this information shall be included in CESSAR.

(2) CENPD-207 - Core Thermo-hydraulics Code

The CENPD-207 is used for the analysis of core thermo-hydraulics and is currently under review by the staff. If final approval of CENPD-207 indicates that any revisions to the analyses are required, this information shall be included in CESSAR.

Our review at this time indicates that there is reasonable assurance that the conclusions based on these analyses will not be appreciably altered by completion of the analytical review. If the final approval of the analytical methods indicates that any revisions to the analyses are required, CESSAR will be required to implement the results of such changes.

Based on previous acceptable analyses for CE plants, on a comparison with other industry models, on independent staff audit calculations, and on previous startup testing experience, we conclude that the analytical methods used are acceptable for the safety analyses performed for the CESSAR design.

15.2 Normal Operation and Anticipated Transients

CE has analyzed several events expected to occur one or more times during the lifetime of the plant. It is demonstrated that all the transients are terminated without exceeding specified fuel design limits (DNBR remains at or

Table 15.1-2 Topical Reports for Codes Used in Safety Analyses

Topical Report	Status
1. <u>Large Break LOCA Code</u>	
CENPD-132	Approved
CENPD-132, Supplement 1	Approved
CENPD-132, Supplement 2	Approved
2. <u>LOCA Blowdown Code</u>	
CENPD-133	Approved
CENPD-133, Supplement 2	Approved
3. <u>LOCA Refill/Reflood Code</u>	
CENPD-134	Approved
CENPD-134, Supplement 1	Approved
4. <u>Fuel Rod Heat Transfer Code</u>	
CENPD-135	Approved
CENPD-135, Supplement 2	Approved
CENPD-135, Supplement 4	Approved
5. <u>Reflood Code When Reflood at Less than 1 Inch per Second</u>	
CENPD-138	Approved
CENPD-138, Supplement 1	Approved
6. <u>Heat Transfer Coefficients for 16 x 16 Fuel Bundles Code</u>	
CENPD-123	Approved
7. <u>Small Break LOCA Evaluation Model Code</u>	
CENPD-137	Approved
CENPD-137, Supplement 1	Approved
8. <u>Reactor Coolant Code for Flow During Coastdown Transient</u>	
CENPD-98	Approved
9. <u>CEA Ejection Analysis Code</u>	
CENPD-188	Approved
CENPD-190	Approved
10. <u>Code Used to Simulate NSSS</u>	
CENPD-107 CENPD-107, Supplement 1 CENPD-107, Supplement 2 CENPD-107, Supplement 3 CENPD-107, Supplement 4 CENPD-107, Supplement 5 CENPD-107, Supplement 6	Under review (approved for code application to ATWS only)

Table 15.1-2 (continued)

Topical Report	Status
11. ATWS Analysis for CE Plants	
CENPD-155	Approved
12. Loss of Flow Analysis Method	
CENPD-183	Approved
13. Core Thermohydraulics Code	
CENPD-161	Approved
CENPD-162	Approved
CENPD-206	Approved
CENPD-207	Under review

above 1.19 using the CE-1 correlation) and that the reactor coolant pressure stays below 110% of system design pressure. For transients with single failure events, core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained. Radiological consequences for the various postulated events are given in Section 15.4 of this report.

15.2.1 Increase in Heat Removal by the Secondary System

CE has analyzed anticipated operational occurrences (AOO), which include increase heat removal by the secondary system as required by the Standard Review Plan Sections 15.1.1, 15.1.2, 15.1.3, and 15.1.4. These sections correspond to the following events:

(1) Decrease in Feedwater Temperature (Section 15.1.1),

(2) Increase in Feedwater Flow (Section 15.1.2),

(3) Increase in Steam Flow (Section 15.1.3), and

(4) Inadvertent Opening of a Steam Generator Relief or Safety Valve (Section 15.1.4).

Sections 15.1.1 thru 15.1.3 were analyzed by a qualitative comparison to the event of an inadvertent opening of a steam generator relief or atmospheric dump valve (SRP Section 15.1.4). The inadvertent opening of an atmospheric dump valve, without an added single failure (i.e., offsite power is available), did not result in a minimum DNBR below 1.19. Therefore, as long as the analyzed conditions for the three AOO events result in less limiting conditions than the open dump valve event, the resulting minimum DNBR would remain above 1.19 and fuel integrity would be maintained. Based on our review, we agree that the inadvertent opening of a steam generator atmospheric dump valve is the limiting event.

CE's analysis of the inadvertent opening of a steam generator atmospheric dump valve event resulted in an 11% increase in main steam flow. This increase in steam flow leads to a 8.9% increase in power demand. Assuming no operator intervention or system malfunctions, this event resulted a core power increase which stabilized at 111% of rated power. The increase in core power resulted from the added steam flow exiting the stuck open dump valve. Since the feedwater control system was assumed to operate on automatic mode, the steam generator water level was maintained and an automatic turbine trip would not be predicted to occur.

CE's analytical assumptions and initial conditions were chosen such that the overpower conditions yielded the limiting approach (DNRB at 1.19) to the specified acceptable fuel design limits (SAFDL) without producing a reactor trip. If the core power increased beyond 111% of rated, the Core protection Calculators (CPC) would initiate a reactor trip and the event would be less severe.

Through comparative analyses of the four overcooling events (identified in the introduction to this section) CE has determined that the inadvertent opening of an atmospheric dump valve resulted in the limiting minimum DNRB (i.e., highest power level). Since this event (without assuming a single failure) did not result in a minimum DNBR of less than 1.19 it was concuded that no fuel damage would occur for any of the four events).

In addition to the AOOs described above, CE assessed the consequence of the limiting single failure for each event. For the decrease in feedwater temperature event, the increase in feedwater flow event and the increase in main steam flow event (i.e., opening of the turbine admission valves), the limiting single failure was stated to be a loss of offsite power resulting from a turbine trip. The inadvertent opening of a steam generator atmospheric dump valve event did not result in a turbine trip, due to the feedwater system maintaining the steam generator water inventory (level). Forty-five seconds into the event, the system reaches a steady-state condition at 111% of design power and a DNBR of 1.19. At this time, loss of offsite power was randomly assumed (not as a result of turbine trip, as assumed for the other three events). Losing offsite power removes electrical energy from the reactor coolant pumps, thus initiating pump coastdown. Due to the resulting decrease in core flow, the core protection calculators initiate a reactor trip signal on low DNBR. After accounting for the appropriate logic delays, the CEAs begin to drop at 46.09 seconds.

During the event described above, the resulting DNBR decreased to 0.93, with 8% of the fuel rods predicted to core experience a DNBR of less than 1.19. These rods were assumed to fail.

When assuming the limiting single failure for the other three remaining AOOs (i.e., loss of offsite as result of turbine trip), fuel damage was also predicted. However, the minimum DNBR was calculated to be greater than 0.93, and thus were not as limiting as the inadvertent opening of the atmospheic dump valve event.

As required by the Standard Review Plan and GDC 27, the transients analyzed assumed the limiting control element assembly (CEA) to be stuck in the withdrawn position throughout the event.

The increase in heat removal events were evaluated using the CESEC-II computer program. CESEC-II is a mathematical computer model that is presently under review by the staff and has been previously utilized by other applicants. The review at this time indicates reasonable assurance that the conclusions based upon the CE submittal will not be appreciably changed at the completion of our review. CE will be required to implement the results of any changes resulting from this review.

CESSAR has demonstrated conformance with the acceptance criteria in Standard Review Plan Sections 15.1.1, 15.1.2, 15.1.3, and 15.1.4. We, therefore, conclude that the CESSAR design is acceptable with respect to accommodating moderate frequency transients resulting in an incrase in heat removal by the secondary system.

15.2.2 Descrease in Heat Removal by the Secondary System

A number of plant transients can result in an unplanned decrease in heat removal by the secondary system. Those that might be expected to occur with moderate frequency are loss of external load, turbine trip, loss of condense vacuum, closure of the main steam isolation valve, loss of normal ac power, and loss of normal feedwater flow. The acceptance criteria for events of moderate frequency are:

(1) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressure.

(2) Fuel cladding integrity should be maintained by ensuring that acceptance Criterion 1 of Standard Review Plan, Section 4.4 is satisfied throughout the transient.

(3) An incident of moderate frequency should not generate a more serious plant condition.

(4) An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding.

CE has analyzed the above events which cause a decrease in secondary side heat removal and identified that the most limiting transient with respect to RCS pressure is the loss of condenser vacuum. For this event, the calculated maximum RCS pressure is 2749 psia. A reactor trip occurs as a result of a high pressurizer pressure trip during the loss of condenser vacuum transient. The most limiting transient with respect to DNBR is the loss of normal ac power where the calculated minimum DNBR is 1.19. A reactor trip occurs as a result of a low DNBR trip during the loss of normal ac power transient.

For the transient combined with a single failure to maximize the peak pressure, no credible failures have been identified which can degrade the pressurizer safety valve capacity. A decrease in RCS to steam generator heat transfer due to reactor coolant flow coastdown can only be caused by a failure to fast transfer to offsite power or a loss of offsite power following turbine trip. These single failures will cause coastdown of the reactor coolant pumps and result in an early reactor trip generated by the core protection calculators on low DNBR. Due to the rapid reactor trip, both of these failures reduced the peak pressure relative to the loss of condenser vacuum event itself. With respect to the fuel performance, no credible single failures have been identified which could result in a more limiting DNBR than that from the loss of normal ac power event itself (a minimum DNBR of 1.19).

CE's calculations show that for transient events leading to decrease in heat removal by the secondary system (with or without single failure), the minimum DNBR is 1.19. Thus, no fuel failure is predicted to occur, core geometry and control rod insertability are maintained with no loss of core cooling capability, and maximum RCS pressure remains below 110% of design. We find the results of these analyses in conformance with the acceptance criteria of the Standard Review Plan 15.2.1 through 15.2.7 and are, therefore, acceptable.

15.2.3 Decrease in Reactor Coolant Flow Rate

CE has analyzed the total loss of forced reactor coolant flow event that leads to a decrease in rector coolant flow. The partial loss of forced reactor coolant flow is bounded by the total loss of forced reactor coolant flow.

A loss of power to all reactor coolant pumps produces a reduction of coolant flow through the reactor core. The reduction in coolant flow rate causes an increase in the average coolant temperature in the core and a decrease in the margin to DNB. A low DNBR reactor trip is generated by the core protection calculators to prevent the minimum DNBR calculated with the CE-1 correlation from decreasing to below 1.19 at any time during the transient. The maximum calculated RCS pressure is 2576 psia during the transient.

For the total loss of forced reactor coolant flow event, the minimum DNBR of 1.19 occurs during the first 3 seconds of the transient, and the reactor is tripped by the CPC's on the approach to the DNBR limit within 1 second of the transient. Any single failure which would result in a lower DNBR during the transient would have to occur during the first 3 seconds after event initiation. CE states that no credible single active failures have been identified that will have any effect on the transient minimum DNBR during this period of time. Therefore, the total loss of forced reactor coolant flow plus a single failure will not result in a lower DNBR than that calculated for the total loss of forced reactor coolant flow event alone. With respect to the peak pressure, the total loss of forced reactor coolant flow plus a single failure event is bounded by the results of a loss of condenser vacuum (LOCV) event. This is because the LOCV has a delayed reactor trip on high pressurizer pressure signal.

CE's analyses show that for transient events leading to a decrease in reactor coolant flow rate (with or without single failure), the minimum DNBR is 1.19. Thus, no fuel failure is predicted to occur, core geometry and control rod insertability are maintained with no loss of core cooling capability, and maximum RCS pressure remains below 110% of design. We find the results of these analyses in conformance with the acceptance criteria of the Standard Review Plan 15.3.1 and 15.3.2 and, therefore, are acceptable.

15.2.4 Reactivity and Power Distribution Anomalies

15.2.4.1 Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition

The consequences of an uncontrolled CEA withdrawal at low power have been analyzed in CESSAR. Such a transient can be caused by a failure in the control element drive mechanism, control element drive mechanism control system, reactor regulating system, or by operator error. The analysis assumes a conservatively small (in absolute magnitude) negative Doppler coefficient and the most positive moderator coefficient. The reactivity insertion rate corresponds to approximately twice the largest insertion rate expected from the sequential withdrawal of the CEA group with 40 percent overlap at the maximum speed of 30 inches per minute. The transient is terminated with a minimum DNBR much greater than 1.19 in the hot channel. Fuel centerline temperatures do not exceed UO_2 melting and the highest RCS pressure produced is well below the emergency limit of 2750 psia.

We have reviewed the reactivity worths and reactivity coefficients used in the analysis and conclude that conservative values have been used. We have reviewed the calculated consequences of this design transient and conclude that they conform with the acceptance criteria in the Standard Review Plan and are, therefore, acceptable.

We, therefore, find that the requirements of GDC 20, which requires that protection be automatically initiated, and GDC 25, which requires that a single failure of the protection system does not result in violation of specified fuel design limits, have been satisfied.

15.2.4.2 Uncontrolled CEA Withdrawal from Power

The consequences of uncontrolled CEA withdrawal in the power operating range have been analyzed in CESSAR. The effect of the resulting power transient causes an increasing temperature and pressure transient which, together with the power distribution shift to the top of the core, causes a rapid approach to the fuel design limits. The initial conditions assumed in the analysis include a power level of 102 percent of full power, a bottom peaked core average axial power distribution, a conservatively small Doppler coefficient, and the most positive moderator coefficient. The CEA withdrawal is initiated from 25 percent insertion of the first regulating bank. The reactivity insertion rate is based on calculated CEA worths and associated uncertainties, and on the maximum withdrawal rate capability of the CEA drive system. The transient is terminated with a minimum DNBR of 1.19 in the hot channel and with fuel temperatures well below centerline melt.

The basis for acceptance in the staff review is that the CE analysis method has been reviewed and approved, the input parameters have been found to be suitably conservative, and the results show that no fuel damage occurs. We conclude that the calculations contain sufficient conservatism with respect to input assumptions and models to assure that fuel damage will not result from control rod withdrawal errors. We, therefore, conclude that the requirements of GDC 20 and 25 have been met.

15.2.4.3 CEA Misoperation Events

The control element assembly misoperation events investigated in CESSAR include individual full-length or part-length CEA drops, and dropping of part-length subgroups. A subgroup is defined as any one set of four symmetrical CEAs which is controlled by the same control element drive mechanism control system.

The effect of any individually misoperated CEA on core power distributions will be evaluated by the CEA calculators, and an appropriate power distribution penalty factor will be transmitted to the core protection calculators (CPCs). The CPCs will, themselves, assess other changes in core conditions (e.g., changes in coolant temperature, axial power distribution, power level) and initiate a low departure from nucleate boiling ratio or high local power density trip if required. However, there are trip delay times associated with the CPC-generated departure from nucleate boiling ratio and high local power density trips, and time is required to insert CEAs following scram. To ensure that the CPCs can accommodate all misoperation events, it must be demonstrated that the elapsed time between initiation of the event and the time the core approaches either the departure from nucleate boiling ratio or local power density limit is sufficient to allow for CPC scram initiation and CEA insertion. Therefore, the misoperation events of most interest are those that result in a rapid decrease in margin to safety limit.

Physics calculations by CE have shown that a CEA withdrawal event is more limiting in DNBR than a part-length CEA subgroup drop since the latter results in a more rapid CPC trip which terminates the event before a core minimum DNBR of 1.19 can be reached. CE has also stated that the drop of a single part-length CEA will not exceed the SAFDL on linear heat rate, and, therefore, on fuel centerline temperature, at any time during the event if the core is operated within the LOCA limits. Therefore, the most limiting CEA misoperation event is the single full-length CEA drop. If the increase in radial peaking factor is large enough, a reactor trip occurs and there is no appreciable decrease in thermal margin. The most limiting CEA misoperation event is the single full-length CEA drop which does not cause a trip to occur but results in an approach to the DNBR criterion of 1.19.

The transient is initiated by the release and subsequent drop of a full-length CEA with a resultant increase in the hot pin radial peaking factor coupled with a return to 102 percent of full power. A minimum DNBR of 1.19 is reached in 36 seconds. The pressure drop beyond this point is arrested by the return to full power and a new steady state is reached at about 50 seconds. The peak centerline fuel temperature obtained during the transient is less than 4000°F. The acceptance guidelines on fuel performance in the Standard Review Plan are, therefore, met.

The analyses of the nuclear steam supply system response were performed using the CESEC-II computer program. The time-dependent thermal margin on DNBR was calculated using the TORC computer program with the CE-1 critical heat flux (CHF) correlation. The sets of initial conditions (power, pressure, temperature, coolant flowrate, radial peaking factors, and axial power distribution) were chosen such that a minimum initial thermal margin was obtained. This was done so that the transient minimum DNBR could be determined as a function of the dropped CEA radial peaking factor increase. This information was then used to select the maximum change in radial peaking factor which, in conjunction with the extreme conditions on other parameters, causes the DNBR to reach 1.19 without a reactor trip.

The staff has reviewed the CEA misoperation events in CESSAR and finds acceptable the general approach used to establish that during the most limiting events, no violations of the specified acceptable fuel design limits on DNBR, centerline fuel temperature, and RCS pressure occur. We conclude that the requirements of GDC 25 have been met, based on conformance with the acceptance criteria in the Standard Review Plan.

15.2.4.4 Startup of an Inactive Reactor Coolant Pump

CE has provided a qualitative analysis for the startup of an inactive reactor coolant pump (SIRCP) event in Section 15.4.4 of the FSAR. This event is not a limiting transient with respect to RCS pressure and fuel performance criteria among the events in the same group category which will result in an increase in core reactivity. The event was evaluated during modes 3 through 6 (hot standby, hot shutdown, cold shutdown, and refueling) since plant operation with less than all 4 reactor coolant pumps is permitted only during those modes of operation. For modes 3 and 4, the primary safety valves, main steam safety valves, and the reactor protection system are designed to maintain the RCS below 110% of design pressure. During modes 5 and 6, when the shutdown cooling system is aligned, overpressure protection is provided by the shutdown cooling system relief valves. For modes 3 and 4, the heat imbalance due to the SIRCP is less limiting than that caused by the CEA withdrawal event. Thus, the maximum RCS pressure will be maintained below 110% of design pressure. In modes 5 and 6, the capacity of the shutdown cooling relief valves prevents the RCS pressure following a SIRCP from exceeding the pressure/temperature limits for these modes.

Regarding the approach to fuel design limits for the SIRCP, the minimum DNBR in the hot channel will increase as the transient progresses. Therefore, no fuel damage is expected.

Based on the above, we find the results of the CESSAR analysis in conformance with the acceptance criteria of Standard Review Plan Sections 15.4.4 and 15.4.5 and therefore, are acceptable.

15.2.4.5 Inadvertent Boron Dilution

Section 15.4.6 of the Standard Review Plan requires that at least 15 minutes is available from the time the operator is made aware of an unplanned boron dilution event to the time a loss of shutdown margin occurs during power operation, startup, hot standby, hot shutdown, and cold shutdown. Thirty minutes warning is required during refueling. CE indicated that operating procedures would be utilized to respond to boron dilution events in modes 3 through 6. The staff has requested that control room alarms be available to alert the operating staff to boron dilution events in all modes of operation. If a second alarm is not provided, CE must show that the consequences of the most limiting unmitigated boron dilution event meet the staff criteria and are acceptable. The staff requires that the applicant provided an analysis for all possible boron dilution events in each of the 6 operational modes and confirm that time intervals which meet the

SRP criteria from the time of the first alarm to the time when the core would go critical is available. Also, technical specifications should be established to restrict when alarms can be taken out of service.

In a letter dated October 29, 1981 CE committed to provide redundant boron alarms for all modes of operation. We will report on the resolution of this issue in a revision to this report.

15.2.4.6 Inadvertent Loading of a Fuel Assembly Into the Improper Position

CESSAR has analyzed a number of postulated interchanges between two fuel assemblies having comparable initial infinite multiplication factors. Because of burnable poison shims in one of the interchanged fuel assemblies, the multiplication factor of that assembly could increase with core burnup. Therefore, the worst fuel loading error undetectable by in-core instrumentation at startup was determined as a function of burnup and a DNB analysis was performed to determine the minimum DNBR. This worst case resulted in a minimum DNBR greater than the minimum acceptable DNBR of 1.19. Since no clad failure is expected to occur under these conditions, expected dose rates are well within the guidelines of 10 CFR 100.

15.2.4.7 Control Element Assembly Ejection

The mechanical failure of a control rod mechanism pressure housing would result in the ejection of a control element assembly (CEA). For CEAs, initially inserted, the consequences would be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Although mechanical provisions have been made to make this accident extremely unlikely, CESSAR has analyzed the consequences of such an event.

Methods used in the analysis are reported in CENPD-190-A, which has been reviewed and accepted by the staff. This report demonstrates that the model used in the accident analysis is conservative relative to a three-dimensional kinetics calculation.

The range of initial conditions examined by CESSAR includes zero power and full power with reactivity coefficients representative of beginning-of-cycle and end-of-cycle for these power level extremes. All cases resulted in a calculated radial average fuel enthalpy for the hottest fuel pellet less than the Regulatory Guide 1.77 acceptance criterion of 280 cal/gm. In addition, the case initiated from full power, initial conditions resulted in the largest number of fuel failures 9.8 percent of the fuel and, therefore, the greatest potential for offsite dose consequences.

For a CEA ejection accident, the staff has traditionally assumed for dose calculational purposes that a fuel rod will fail if its DNBR falls below the approved DNBR limit value. CESSAR proposes to assume that the number of failed fuel rods equals the number of rods in DNB as calculated with the statistical convolution method described in CENPD-183. That is, since the probability of occurrence of DNB is a function of the DNBR, the statistical convolution technique involves the summation over the reactor core of the number of rods with a specific DNBR times the probability of DNB at that DNBR. We have reviewed this model and conclude that it will provide a conservative method for calculating the number of rods in DNB for the CEA ejection analysis for CESSAR and is, therefore, acceptable.

The predominant failure mechanism for CEA ejection accidents is expected to be pellet-to-cladding mechanical interaction (PCI). PCI is not a function of DNB, but is related to the energy inserted into the rods. Such failures are, therefore, correlatable with the fuel rod enthalpy during the transient. For that reason, at the request of the staff, the applicant presented additional information on the total radially averaged enthalpy in the fuel, including a census of the number of fuel pins above a given radially averaged fuel pellet enthalpy. No fuel rods were determined to have radially averaged peak fuel enthalpies above values that appear to result in cladding failures (based on results from a limited number of experimental tests described in Reference 4) as compared to the 9.8 percent calculated by the DNB convolution method. The sensitivity study the staff requested that CE provide, using an energy deposition analysis method, confirmed that the DNB convolution method for calculating fuel failures for the CEA ejection analysis is appropriate.

The ejected rod worths and reactivity coefficients used in the analysis have been reviewed and have been judged to be conservative. The assumptions and methods of analysis used by CE are also in accordance with those recommended in Regulatory Guide 1.77. Therefore, we conclude that this analysis is acceptable.

15.2.5 Increase in Reactor Coolant System Inventory

CESSAR has evaluated events resulting in an increase in reactor coolant system inventory, and has identified the limiting event to be the pressurizer level control system (PLCS) malfunction in combination with the loss of offsite power as a result of the assumed grid failure following the turbine trip. CE has evaluated possible single failures in this analysis and has concluded that none of the credible single failures will result in a higher RCS pressure than that predicted for a PLCS malfunction with a loss offsite power as a result of turbine trip. This event is more limiting than the inadvertant operation of the emergency core cooling system because the shutoff head of the high pressure safety injection pumps is less than the RCS pressure during power operation.

For the limiting event of RCS inventory increase, CE assumes that when the pressurizer level controller fails low or the level setpoint generated by the reactor regulating system fails high, a low level signal can be transmitted to the controller. In response, the controller will strat all 3 charging pumps and close the letdown control valve to its minimum opening resulting in the maximum mass addition to the RCS.

The increase in RCS inventory results in a pressurizer pressure increase to the high pressure trip set point and trips the reactor. Pressurizer safety valves open 2 seconds after the reactor trip and the calculated maximum RCS pressure during this transient is 2561 psia. The 796 lbs of steam calculated to be discharged through the pressurizer safety valves is contained in the quench tank with no releases to the atmosphere. Since this transient causes an increase in RCS pressure due to an increase in reactor coolant inventory, the DNBR increases from the initial conditions. Therefore, the acceptance criterion regarding fuel performance is met, because no fuel failures would be expected.

Based on the above, we find the results of the CESSAR analysis in conformance with the acceptance criteria of the Standard Review Plan Sections 15.5.1 and

15.4.5 with respect to peak RCS pressure and fuel performance and, therefore, is acceptable.

15.2.6 Conclusions

CESSAR has presented results for various anticipated operational occurrences (with and without assumed single failures). With the exceptions noted below, the staff finds they meet NRC acceptance criteria with respect to fuel and primary system performance. Therefore, the applicant has provided adequate protection for anticipated operational occurrences (except as noted) and is considered in compliance with GDC 10, 15, and 26.

In the inadvertent boron dilution event the staff will confirm that CESSAR provides redundant alarms to positively identify boron dilution events in all modes of operation and provides analysis for boron dilution events in each mode of plant operation.

15.3 Limiting Accidents

CESSAR has analyzed events, which, though not expected to occur during the lifetime of the plant, could have serious radiological consequences if not effectively mitigated. For accident conditions, the reactor coolant pressure should stay below the applicable ASME Code limits. The core geometry should be maintained so that there is no loss of core cooling capability and control rod insertability. Radiological consequences are discussed in detail in Section 15.4 of this report.

15.3.1 Steam Piping Failures Inside and Outside of Containment

The staff has reviewed the CESSAR evaluation of postulated steam line breaks (SLBs). The effects of a steam line break can be catagorized as either a limiting pre-reactor trip event or a limiting post-reactor trip event. Large break areas result in a rapid depressurization of the secondary system. These breaks rapidly lead to a reactor trip on low steam generator secondary system pressure. As such, these breaks are typically post-trip limiting events. Depending on the severity of the overcooling of the primary system, a post-trip return to criticality can result from the feedback of the negative moderator and Doppler reactivities.

Small steam line breaks can be catagorized as limiting pre-reactor trip events. These events result in a power increse as the break flow, added to the normal steam flow, provides an added power demand to the reactor core. As in the case for the inadvertent opening of an atmospheric dump valve event (see Section 15.2.1), the primary system power can attain a steady-state value above the rated core power. For break areas which result in an excess of 11% of rated steam flow, the power excursion is expected to inititate a reactor trip on a high power (ΔT) or a low DNBR trip.

The following describes the CESSAR conformance to the Standard Review Plan Section 15.1.5, which requires the evaluation of both the large and small steam line break events.

Large Steam Line Breaks (Post-Reactor Trip Events)

CE conducted four double-ended guillotine steam line break analyses in order to determine the limiting assumptions for post-reactor trip return to power, and tc assess the radiological consequences. The four cases analyzed for post-trip return to power were:

(1) A large steam line break during full power operation in combination with a single failure, loss of offsite power, and a stuck CEA.

(2) A large steam line break during full power operation in combination with a single failure, offsite power available, and a stuck CEA.

(3) A large steam line break during zero power operation in combination with single failure, loss of offsite power, and a stuck CEA.

(4) A large steam line break during zero power operation in combination with a single failure, offsite power available, and a stuck CEA.

The severity of a large steam line break is limited by an integral flow restrictor which has been designed within each steam generator outlet nozzle. The flow restrictor limits the blowdown area to 1.28 square feet. This is equivalent to approximately 30% of steam line cross-sectional area.

CE has conducted a parametric study to assess the limiting single failure for a postualted steam line break (SLB). It was determined that the limiting single failure which could occur during a steam line break with concurrent loss of offsite power (i.e., Cases 1 and 3 above) was the failure of one of the high pressure safety injection (HPSI) pumps to start following a safety injection actuation signal (SIAS). For SLB events with offsite power available (Cases 2 and 4), the limiting single failure was determined to result from a failure to close a main steam isolation valve (MSIV) on one of the steam lines on the intact generator following a main steam isolation signal (MSIS). Cosnequently, steam continues to be released fromt eh intact steam generator after MSIS at a maximum rate of 11% of design steam flow. This steam flow is limited by the cross sectional area of the communicative lines between the intact and broken steam lines. This open flow path is represented by an effective flow area of 0.2556 square feet.

For all four cases analyzed above, the minimum departure from nucleate boiling ratio (DNBR) did not decrease below 1.19. Cases 2 and 4 (SLBs at full and zero power, with offsite power available) did not result in a return to criticality. The respective maximum total reactivity for these cases were -0.01% $\Delta\rho$ and -0.1% $\Delta\rho$.

Cases 1 and 3 (SLBs at full and zero power, with concurrent loss of offsite power) did result in a return to criticality. However, the return to criticality was not sufficient to result in any predicted fuel failure (i.e., the minimum departure from nucleate boiling ratio remained greater than 1.19). The respective total reactivity for these events were +0.004% $\Delta\rho$ and +0.001% $\Delta\rho$.

A main reason that a SLB event with loss of offsite power results in a positive total reactivity is due to the analytical assumptions regarding the treatment of the moderator feedback. CESSAR utilizes the broken steam generator outlet temperature (i.e, cold leg temperature) for the reactivity calculation. For events with offsite power available, CE provided analyses which assumed the reactor coolant pumps remained operational throughout the event. CE also assessed the implications of manually tripping the reactor coolant pumps as result of an ECC initiation signal, as stipulated in the operator guidelines. This assessment will be formally documented in Appendix 15C of the CESSAR FSAR and will indicate that the consequences of tripping the reactor coolant pumps during the event is bounded by the analysis which assumed loss of offsite power at time of break initi For the events with offsite power not available, the reactor coolant pumps coast down, leading to degraded core flow (natural circulation). During this event, the time of primary coolant residency within the broken steam generatory is extended. As a result, more energy (heat) is removed for every pound of coolant which passes through the broken steam generator. This leads to cooler steam generator exit temperatures. This lower primary coolant temperature results in a more positive moderator reactivity feedback.

The limiting large steam line break event analyzed for assessing the radiological consequences at the site exclusion area boundary was determined to be a steam line break outside containment, upstream of the MSIV at zero power operation and in combination with a loss of one high pressure injection pump. This case is identical to Case 3 above, with the break located outside containment. The radiological consequences for this event are more limiting than the corresponding full power events because of the increased coolant mass inventory within the secondary system.

Small Steam Line Breaks (Pre-Reactor Trip Events)

CESSAR FSAR concludes that the limiting steam line break size which produced the greatest challenge to the fuel integrity resulted from a double-ended guillotine break. This break size did not result in predicted fuel failure, since the minimum departure from nucleate boiling ratio (DNBR) did not decrease below the minimum limit of 1.19. However, the staff requested that CE verify the FSAR conclusions by providing confirmatory small break analyses, performed in accordance with the criteria of Standard Review Plan Section 15.1.5. These analyses should document the influences resulting from losing offsite power at time of break initiation; maintaining offsite power throughout the transient; and tripping of the reactor coolant pumps in accordance with operator procedures, should the appropriate conditions exist.

Based upon the results presented in Section 15.1.4 of CESSAR (the inadvertent opening of an atmospheric dump valve event with offsite power available), the staff has reasonable assurance that the small steam line break events will not result in predicted fuel failure. Conditional upon the confirmatory analyses which will demonstrate the limiting nature of large breaks, we find the CESSAR analyses and conclusions acceptable.

Analytical Methodology

The steam line break analyses were originally evaluated by the application of the CESEC-II computer program. CESEC-II is a mathematical computer model presently under review by the staff. During the course of review, the staff questioned the code's ability to adequately assess the primary system response to overcooling events, which resulted in voiding of the upper head. This reduced the operator's ability to control primary system pressure. CE responded to the staff concerns by reanalyzing these events with the CESEC-III computer program. In addition, CE provided documentation and verification of the CESEC-III computer model. The model verification included the analysis of a St. Lucie 1 transient.

In addition to the St. Lucie 1 event, the staff questioned CE on the asymmetric hydraulic behavior during a postulated steam line break. Specifically, the CESEC-II computer program homogenized the primary coolant temperature between the broken and the intact loops as the coolant entered the reactor vessel.

The CESEC-II calculated hot leg temperatures indicated identical temperatures in each hot leg versus time. CE assessed the asymmetric reactor vessel response by reanalyzing the event with the CESEC-III computer program, which simulates incomplete core mixing. Asymmetric analyses (specifically conducted for St. Lucie Unit 2) indicated a potential for a hot leg temperature differential of 100°F. In order to address the staff's concerns, CE reanalyzed the steam line break events with the CESEC-III computer program. These reanalyses have been incorporated into the CESSAR FSAR.

The staff is undergoing a detailed review of the CESEC-III computer programs. Draft documentation of the CESEC-III program has been submitted by CE while final documentation is being completed. As part of the review of the CESEC codes, the staff will conduct a confirmatory audit of CESSAR. CE has committed to provide input parameters needed to develop a computer model of System 80.

In addition to the limiting steam line break event, CE documented the methodology for conforming with the Standard Review Plan Section 15.1.5. Appendix C of the CESSAR FSAR addressed these assumptions. In addition, Appendix C also documented the methodology for consideration of the worst single active failure. Some of the active component failures reviewed included:

(1) Failure of one HPSI pump to start after SIAS

(2) Failure of one main feedwater isolation valve to close after MSIS

(3) Failure of one main steam isolation valve to close after MSIS

(4) Failure of the turbine stop valve to close after reactor trip

(5) Failure of one diesel generator to start after the loss of offsite power

The limiting single active component failure was determined to be the failure of one HPSI pump to start. This delayed the time for the injected boron to reach the reactor core, thus resulting in a higher maximum post-CEA insertion reactivity.

Appendix C documented some of the conservative assumptions incorporated into the parameters used during the analysis of a large steam line break. These include:

(1) No moisture carry over during the steam generator blowdown

(2) Increase in the slope of the Doppler reactivity by 25%

(3) Increase in the slope of the moderator reactivity coeficient by 10%

(4) The moderator feedback was based upon the cold leg temperature of the broken loop steam generator

The reactivity inputs were modified from the end of cycle design values.

As required by the Standard Review Plan Section 15 1.5, the limiting CEA was assumed stuck in the fully withdrawn position after reactor trip.

Our review of the CE computer codes indicate reasonable assurance that the conclusions will not be appreciably changed by the completion of the code review. Should our review indicate that revision to the analyses are necessary, CE will be required to revise the steam line break analyses as appropriate.

Evaluation Findings

In order to demonstrate compliance to Standard Review Plan Section 15.1.5, regarding postulated steam line break events, CE addressed three areas of concerns. These are:

(1) Consequences of large steam line breaks

(2) Consequences of small steam line breaks

(3) Analytical methodology used in assessing the consequences of steam line break events.

The following describes our conclusions to these concerns.

(1) Large Steam Line Breaks

- CE has demonstrated the adequacy of CESSAR to withstand postulated large steam line breaks. During these events, the primary system was not predicted to result in breaching the fuel integrity.

- For the events which assumed offsite power availability, CE assumed continuous running of the reactor coolant pumps. Operator guidelines require the tripping of the reactor coolant pumps on ECC initiation. CE has assessed (but not yet documented) the consequences of the operator tripping the pumps and has determined that the consequences are bounded by the event which assumed loss of offsite power during break initiation. CE has agreed to document these results in Appendix 15C of the CESSAR FSAR.

(2) Small Steam Line Breaks

- CE concluded that the limiting consequences resulting from a steam line break event will occur for postulated large break areas. The staff requested that CE verify the CESSAR conclusions by providing confirmatory small break analyses, performed in accordance with the criteria of SRP Section 15.1.5. These analyses should document the consequences resulting from losing offsite power at time of break initiation; maintaining offsite power throughout the event; and tripping of the reactor coolant pumps in accordance with operator procedures, should the conditions exist.

- Based on the results presented in Section 15.1.4, we have reasonable assurance that the small steam line break will not result in predicted fuel failures for the analyses listed above.

(3) Analytical Methodology

- CE utilized the CESEC-III computer model for analyzing steam line break events. This model is undergoing staff review. Our review at this time indicates reasonable assurance that the conclusions based on the CE submittal will not be appreciably changed by completion of review.

- As an overall independent assessment of the CESSAR submittal, the staff will conduct confirmatory audits of selected Chapter 15 events. CE has committed to supply the required data necessary for this assessment, which will include the steam line break events.

Conditional upon confirmation of the events listed above, we conclude that the consequences of postulated steam line breaks meet the requirements set forth in GDC 27, 28, 31, and 35 regarding control rod insertability and core coolability. This conclusion is based on the following:

(a) CESSAR has met the requirements of GDC 27 and 28 by demonstrating that no fuel damage resulted during the course of the event, control rod insertability would be maintained, and that no loss of core cooling capability resulted.

(b) CESSAR has met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(c) The analyses and effects of steam line break accidents inside and outside containment, during various modes of operation and with and without offsite power, have been reviewed and evaluated using a mathematical model that is under staff review. However, our review at this time indicates reasonable assurance that the conclusions based on the licensee's submittal will not be appreciably changed by completion of review.

(d) The parameters used as input to this model were reviewed and found to be suitably conservative.

(e) CESSAR has met the requirements of TMI Action Plan (see Section 22) Item II.E.1.2, with respect to demonstrating the adequacy of the auxiliary feedwater design to remove decay heat following feedwater piping failures.

(f) CE assessed the implications of manually tripping the reactor coolant pumps as a result of an ECC initiation signal, as required by Task Action Plan (see Section 22) Item II.K.3.5. The assumptions used are conservative. The staff has reasonable assurance that the results of the CESSAR analyses will be consistent with the generic resolution to Item II.K.3.5.

15.3.2 Feedwater System Pipe Breaks

Section 15.2.8 of CESSAR referenced FSAR Appendix 15B for the feedwater line break analysis. Appendix 15B addresses both the consequences and the analytical methodology used in analyzing postulated feedwater line breaks occurring between the steam generator feedwater nozzle and the reverse flow check valve, upstream of the nozzle. In addition, Appendix 15B also documents the analytical methodology including detailed sensitivity studies conducted to derive at the limiting initial operating conditions.

Feedwater line breaks are analyzed as pressurization events. Since the minimum departure from nucleate boiling ratio (DNBR) did not decrease below the initial conditions (thereby remaining well above the 1.19 limit for predicting fuel failure), it is concluded that the fuel integrity was maintained throughout the event.

During a feedwater line break, the enthalpy exiting the break can influence the severity of the event. High quality break flow will tend to reduce the primary system pressure as it enhances the cooling of the secondary system. Low enthalpy break flow, on the other hand, will minimize the secondary system depressurization and therefore result in higher primary system pressures. In order to bound the effects of break flow conditions, CESSAR has conservatively assumed zero quality break flow throughout the blowdown of the broken steam generator.

CE has determined that the primary system peak pressure is a strong function of the rate of heat transfer degradation between the primary and secondary systems. Due to the simplistic representation of the secondary system modeled in the CE computer program, the limiting influence of the heat transfer degradation rate was assessed by conducting a series of sensitivity studies. These studies linearly degraded the heat transfer as a function of remaining liquid inventory in the broken steam generator. As an example, Figure 15B-1 in the FSAR documented the sensitivity of the primary system peak pressure for heat transfer degradation beginning at a secondary system liquid inventory of 100,000, 60,000, 30,000, and 0 lbm. A degradation of the heat transfer at zero liquid inventory results in a step decrease of heat transfer between the primary and secondary systems. This, in turn, results in the largest imbalance of primary-to-secondary heat transfer removal rate. As a result, the system pressurization rate was maximized.

During the duration of the secondary side blowdown, CE conservatively did not credit a reactor trip on low steam generator level until the entire liquid inventory was depleted from the broken steam generator. Nor was credit taken for a high containment pressure trip, should that have occurred before the inventory of the broken generator emptied. Instead, CE bounded the analysis by only crediting a high pressurizer pressure trip and a low steam generator level trip at the time the broken steam generator liquid inventory was totally depleted.

Utilizing the assumptions outlined above, CESSAR analyzed a spectrum of break sizes and concluded that the peak primary system pressure occurred for a 0.2 ft^2 break. A double-ended guillotine break would result in an effective break area of 1.4 ft^2. For the 0.2 ft^2 event, the primary system reached a peak pressure of 2843 psia. This pressure exceeds 110% of the primary system design pressure, and does not conform to the criteria of Standard Review Plan Section 15.2.8, which limits the system pressurization to 110% of design pressure. The Standard Review Plan established its limit based on the ASME

Code. The staff has considered the CESSAR analyses and has concluded that the ASME Service Level C pressure limit is more appropriate for large breaks within the feedwater system, including small breaks which are accompanied by loss of offsite power. Crediting Service Level C limits results in a new pressure limit of approximately 120% of the system design pressure. This limit is still within the elastic strength of the material properties of the system. We, therefore, conclude that the results of the CESSAR analysis for this event are acceptable.

Service Level C pressure limits are not considered appropriate by the staff for small feedwater line breaks with offsite power available. We, therefore, require that CE provide confirmatory analyses demonstrating that a small feedwater line break (with the limiting single failure and offsite power available) will not result in exceeding 110% of the design system pressure, as required by SRP Section 15.2.8.

CE provided detailed justification for the initial conditions assumed in the CESSAR analyses. These included detailed sensitivity studies of the following input parameters: initial reactor coolant system pressure; initial core power; initial reactor vessel flow; initial pressurizer water volume; pressurizer safety valve rated flow uncertainties; Doppler multiplier; core life; internal stored energy within the fuel; initial steam generator inventory; initial feedwater enthalpy; and initial core inlet temperature. Results of these sensitivity studies provide a set of initial conditions and transient parameters which establish the limiting RCS overpressurization event. In summary, this set includes:

(1) 0.2 ft^2 break area.

(2) Instantaneous loss of heat transfer in the ruptured steam generator.

(3) Initial RCS pressure which forces a high pressurizer pressure trip coincident with the first reactor trip signal.

(4) Nominal reactor vessel flow.

(5) Maximum initial core power.

(6) Maximum initial pressurizer liquid volume.

(7) Minimum pressurizer safety valve rated flow.

(8) Nominal Doppler reactivity feedback.

(9) Most positive moderator temperature coefficient of reactivity.

(10) Minimum fuel gas gap heat transfer coefficient.

(11) Nominal initial steam generator water mass.

(12) Minimum initial feedwater enthalpy.

(13) Maximum initial core inlet temperature.

The staff concurs with the methodology utilized in arriving at the limiting input parameters and finds these conditions acceptable for analyzing feedwater line breaks.

The Standard Review Plan requires consideration of the worst single failure during this event. CESSAR has not addressed the worst possible single failure. However, CE has stated that a review was conducted to assess a limiting failure which could further degrade the primary system. No additional single failure could be identified CE by to further degrade the system. The staff will require formal documentation of these conclusions and their bases.

We intend to continue our evaluation of the modeling assumptions made in the CESSAR analyses of a feedwater line break. Should the results of this evaluation indicate the assumptions made to be inappropriate, we will require that CE modify the analysis model accordingly and to reanalyze the event.

As required by the Standard Review Plan, the feedwater line break event was analyzed assuming the limiting control element assembly (CEA) to be stuck in the withdrawn position throughout the event.

The feedwater line break event was analyzed with the CESEC-II computer program. CESEC-II is a mathematical computer model that is presently under review by the staff and has been previously utilized on other CE plants. The review results at this time indicate reasonable assurance that the conclusions will not be appreciably changed at the completion of review. CE will be required to implement the results of any changes resulting from these reviews.

Conditional upon confirmation that a small feedwater line break event, with offsite power available, including the limiting single failure, will not exceed 110% of the design system pressure, as well as documentation confirming that no single failure will result in a more limiting pressurization, we conclude that the consequences of postulated feedwater line breaks meet the requirements set forth in GDC 27, 28, 31, and 35 regarding control rod insertability and core coolability. This conclusion based on the following:

(a) CESSAR has met the requirements of GDC 27 and 28 by demonstrating that no fuel damage resulted during the course of the event, control rod insertability would be maintained, and that no loss of core cooling capability resulted.

(b) CESSAR has met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(c) The analyses and effects of feedwater line break accidents inside and outside containment, during various modes of operation and with and without offsite power, have been reviewed and evaluated using a mathematical model that is under staff review. However, our review at this time indicates reasonable assurance that the conclusions will not be appreciably changed at completion of review.

(d) The parameters used as input to this model were reviewed and found to be suitably conservative.

(e) The applicant met the requirements of TMI Action Plan (see Section 22) Item II.E.1.2 with respect to demonstrating the adequacy of the auxiliary feedwater design to remove decay heat following feedwater piping failures.

15.3.3 Single Reactor Coolant Pump Shaft Seizure

During a reactor coolant pump shaft seizure accident, the following criteria must be met. Core geometry should remain intact so that there is no loss of core cooling capability or control rod insertability. Loss of offsite power and the Technical Specification limit for steam generator tube leakage should also be assumed in the analyses for this event. Reactor coolant pressure should be maintained below 110% of design pressure, and a rotor seizure, by itself, should not degenerate into a more serious condition or result in the loss of function of the RCS or containment barriers. Radiological consequences are discussed in Section 15.4 of this report.

CE is currently reanalyzing this event with regard to the radioilogical consequences, as discussed in Section 15.4.2. Based on our review thus far, we have reasonable assurance that the fuel performance and peak system pressure of the new analysis will be acceptable. We will report on the resolution of this issue in a revision to this report.

15.3.4 Single Reactor Coolant Pump Shaft Break

During a reactor coolant pump shaft break accident, the acceptance criteria are the same as the criteria stated in Section 15.3.3.1 for the reactor coolant pump shaft seizure accident.

For the pump shaft break accident, the reactor is tripped on differential pressure across either steam generator, where as for the pump shaft seizure accident the reactor is tripped by the CPC on a low projected DNB condition. The flow coastdown for a pump shaft seizure accident is faster than the coastdown for a pump shaft break accident. Although the reactor trip time is approximately 0.3 seconds later for the pump shaft break accident, the minimum DNBR is no less than that of the pump shaft seizure accident. This is the result of a less severe flow coastdown in the shaft break analysis.

We have reviewed the CESSAR qualitative analyses for the event and agree that the results of the reactor coolant pump shaft break are bounded by that of the reactor coolant pump seizure accident addressed in Section 15.3.3.1 of this report. Therefore, the results are acceptable.

15.3.5 Inadvertent Opening of a Pressurizer Safety or Relief Valve

The design of the CESSAR reactor coolant system (RCS) includes four safety valves on the pressurizer. The RCS does not have power-operated relief valves. CE has analyzed the inadvertent opening of a pressurizer safety valve using their small-break LOCA model. A low pressurizer pressure signal initiates reactor trip and turbine trip. A loss of offsite power is assumed to occur simultaneously with reactor trip. The worst single failure was identified as failure of one diesel to start. Thus, only one HPSI and one LPSI pump are assumed to be available. The results of the calculation show that the core does not uncover and the peak cladding temperature is 1012°F.

The intent of SRP Section 15.6.1 is to address those valves which have control systems that may fail causing an inadvertent opening (e.g., a pressurizer relief valve). The safety valves are considered passive devices. Thus, this event is not evaluated against the performance criteria for anticipated operational occurrences, but rather should meet the criteria of 10 CFR 50.46. Since the peak cladding temperature remained well below the 2200°F limit, the results are considered acceptable. The results of this analysis are bounded by the small-break LOCA spectrum, as presented in Section 15.3.8 below.

15.3.6 Double-Ended Break of a Letdown Line Outside Containment

Direct release of reactor coolant may result from a break or leak outside of containment in a letdown line, instrument line, or sample line. The double-ended break of the letdown line outside containment, upstream of the letdown line control valve was selected for this analysis because it is the largest line and results in the largest release of reactor coolant outside the containment.

A double-ended break of the letdown line outside containment, upstream of the letdown line control valve, releases primary fluid to the auxiliary building at a rate of approximately 50 lbs/sec. This is more than twice the maximum expected letdown flow. The event will actuate a number of alarms that would be noted by the reactor operator in the control room. The first three, that is, the RHX exit high temperature alarm, the letdown line low flow and low pressure alarms, and the low flow alarms in the process radiation monitor and the boronometer, are going to immediately alert the operator after the initiation of the event.

The analysis assumes that 10 minutes after the first of three alarms resulting from the event, the operator isolates the letdown line thereby terminating any further release of primary flow to the auxiliary building. Subsequently, the operator is assumed to take appropriate steps for a controlled reactor shutdown. The assumption of operator action within 10 minutes after the first few alarms are triggered is consistent with the criteria set forth in ANS 58.8, ANSI N660, Rev. 2, 1981 ("Time Response Design Criteria for Safety-Related Operator Actions"). This is the minimum time for the letdown line break event category that shall elapse from the time of the alarm until operator actions can be considered for initiation of safety functions.

The CESEC-II computer program was used to simulate the event. CESEC-II does not account for void formation in the primary system, as might be expected for this event if no operator actions were taken. Since no void formation occurs during the first 10 minutes of the transient, the use of CESEC-II for this event is acceptable.

The double-ended break of a letdown line outside containment upstream of the letdown line control valve results in gradual depressurization of the reactor coolant system. The minimum departure from nucleate boiling ratio (DNBR) stays above the initial value of 1.65 throughout the transient. Hence, no fuel pins are calculated to experience DNB for this event.

During the 600-second duration of the transient no more than 30,766 lbs of primary system coolant is released outside the containment.

The staff finds the assumptions used and the analysis performed for this event to be acceptable and that the scenario, as described in CESSAR, assures that the most severe failure of a small line carrying primary coolant outside containment has been considered.

15.3.7 Steam Generator Tube Rupture

The steam generator tube rupture (SGTR) accident is a penetration of the barrier between the reactor coolant system (RCS) and the main steam system and results from the failure of a steam generator U-tube. Integrity of the barrier between the RCS and main steam system is significant from a radiological release standpoint. The radioactivity from the leaking steam generator tube mixes with the shell-side water in the affected steam generator. Prior to turbine trip, the radioactivity is transported through the turbine to the condenser where the noncondensible radioactive materials would be released via the condenser air ejectors. Following reactor trip and turbine trip, with the steam bypass system in its manual model, the steam generator safety valves open to control the main steam system pressure. The operator can isolate the damaged steam generator any time after reactor trip occurs. The cooldown of the NSSS can then be performed by manual operation of the emergency feedwater and the steam bypass control system (SBCS), and using the unaffected steam generator. The analysis presented in CESSAR conservatively assumes that operator action is delayed until 30 minutes after the initiation of the event.

The radiological consequences for the SGTR transient, which are evaluated in Section 15.4.5, are also dependent on the break size. For break sizes resulting in a reactor trip during the first 30 minutes of the accident, the initial leak rate decreases from that value equivalent to a double-ended rupture, and the offsite dose also decreases due to the drop in the integrated leak. The decrease in break size also delays the time of reactor trip. As the break size is decreased further, the integral leak is reduced for the 30-minute operator action interval and the radiological consequences will be less severe. Therefore, the most adverse break size is the largest assumed break of a full double-ended rupture of a steam generator tube.

The CESEC-III computer program was used to simulate the SGTR event. CESEC-III accounts for void formation in the primary system once the pressurizer empties. The SGTR event is the most limiting event with respect to void formation.

Voids form in the reactor vessel upper head region during the accident, due to the thermal hydraulic decoupling of this region from the rest of the RCS following RCP trip. The upper head region liquid level remains well above the top of the hot leg throughout the transient. Furthermore, the upper head voids begin to collapse upon actuation of the safety injection flow. After 30 minutes, the operator employs the plant emergency procedure for the steam generator tube rupture event to cooldown the plant to shutdown cooling entry conditions.

The SGTR event was analyzed both with and without a concurrent loss of offsite power (LOP) at the time of reactor trip. A limit of 1 gpm leakage in the unaffected steam generator was assumed for the duration of the transient.

The maximum RCS and secondary pressure do not exceed 110% of design pressure following a steam generator tube rupture event without concurrent loss of offsite power, thus assuring the integrity of the RCS and main steam system.

The minimum DNBR of 1.22 is above the minimum DNBR limit of 1.19 and therefore no fuel failure is assumed to occur.

The maximum RCS and secondary pressures do not exceed 110% of design pressure following a steam generator tube rupture event with a concurrent loss of offsite power, thus assuring the integrity of the RCS and the main steam system. The minimum DNBR of 1.13 is only slightly lower than the 1.19 limit. The number of fuel pins calculated to experience DNB during the transient is negligible (0.32%). All pins experiencing DNB are conservatively assumed to fail.

The plant is maintained in a stable condition due to automatic actions, and after 30 minutes, the operator employs the plant emergency procedure for the steam generator tube rupture event to cooldown the plant to shutdown cooling entry conditions.

The staff finds the assumptions used and the analyses performed for this event to be acceptable and that the scenarios, as described in CESSAR, assure that the most severe SGTR event has been considered.

15.3.8 Loss-of-Coolant Accident (LOCA)

The acceptance criteria for a LOCA as required by 10 CFR 50.46 are:

(1) The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(2) The calculated total oxidation of the cladding shall not exceed 17% of the total cladding thickness before oxidation.

(3) The calculated total amount of H_2 generated from the chemical reaction of the cladding with water or steam shall not exceed 1% of the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core shall remain amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed for the extended period of time required by long-lived radioactivity.

The details of the ECCS mitigating and long-term cooling systems for a LOCA are provided in Section 6.3 of this report. The calculations are made using approved computer programs and models which meet the requirements of Appendix K to 10 CFR Part 50. The initial conditions are chosen to maximize the cladding temperature and oxidation. Containment parameters are chosen to minimize the calculated containment pressure to assure that the reflood calculations are conservatively calculated. During the LOCA calculation, offsite power is assumed to be lost.

CESSAR has analyzed a complete break spectrum for the large-break LOCA (1.0, 0.8, and 0.6 double-ended slot and guillotine breaks in the pump discharge leg, a 1.0 double-ended guillotine break in the pump suction leg and in the hot leg, and a 0.5 ft^2 slot break in the pump discharge leg) in Section 6.3.3.2 of the FSAR.

The time of ECCS pumped flow delivery to the core includes a delay time for the startup of the diesel generators. Studies show that the worst single failure for the large-break LOCA spectrum analyses is the failure of one low pressure injection pump to start. All ECCS flow delivery, both pumped and accumulator injection, to the broken cold leg is assumed to spill directly into containment. CE performed cladding ballooning calculations which showed that none of the LOCAs analyzed had core geometry changes of a magnitude large enough to significantly reduce core cooling capabilities. The calculations for core geometry were carried out past the point where temperatures were decreasing and the primary system had depressurized.

The large-break spectrum analyses result in the following:

(1) The peak calculated cladding temperature is 2169°F for the 1.0 double-ended guillotine break in the pump discharge leg.

(2) The maximum local cladding oxidation is calculated to be 13.32% for the 1.0 double-ended slot break in the pump discharge leg.

(3) The maximum core-wide cladding oxidation is calculated to be 0.799% for the 1.0 double-ended slot break in the pump discharge leg.

The small-break LOCA spectrum was provided in Amendment 4, Section 6.3.3.3, to the CESSAR FSAR. CE analyzed a complete small-break LOCA spectrum (0.5 ft^2, 0.35 ft^2, 0.20 ft^2, 0.05 ft^2, and 0.02 ft^2 breaks in the pump discharge leg, an 0.03 ft^2 break, equivalent to the flow area of a pressurizer safety valve in the pressurizer, and an 0.003 ft^2 break in the reactor vessel lower plenum, to simulate the failure of an instrument tube).

Studies show that the worst single failure for a small-break LOCA is the failure of one diesel to start, resulting in the loss of one high pressure pump and one low pressure pump. All ECCS flow delivery, both pumped and accumulator injection, to the broken cold leg is assumed to spill directly to containment.

The small-break spectrum analyses result in the following:

(1) The peak calculated cladding temperature is 1557°F for the 0.05 ft^2 pump discharge leg break.

(2) The maximum local cladding oxidation is calculated to be less than 0.8825% for the 0.05 ft^2 pump discharge leg break.

(3) The maximum core-wide cladding oxidation is calculated to be less than 0.143% for the 0.05 ft^2 pump discharge leg break.

The CESSAR ECCS analysis does not assume any steam generator tubes are plugged. The effects of tube plugging is treated on an "as needed" basis for CE operating plants. A sensitivity analysis assuming 6% plugging showed minimal changes in the ECCS performance and no change in the allowable peak linear heat generation rate. Based on this, the current ECCS performance analysis, which does not consider steam generator tube plugging, is acceptable and no new analysis is required unless the plugging in CESSAR plants exceeds 6%.

We conclude that the loss-of-coolant analysis resulting from a spectrum of postulated piping breaks within the primary coolant pressure boundary is acceptable and meets the relevant requirements of 10 CFR Part 50.46 and Appendix K to Part 50. This conclusion is based on the following:

CE has performed analyses of the performance of the emergency core cooling system (ECCS) in accordance with the Commission's regulations (10 CFR 50.46 and Appendix K to Part 50). The analyses considered a spectrum of postulated break sizes and locations and were performed with an evaluation model which had been previously reviewed and approved by the staff. The results of the analyses show that the ECCS satisfies the following criteria:

(1) The calculated maximum fuel rod cladding temperature does not exceed 2200°F.

(2) The calculated maximum local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptable low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

15.3.9 Anticipated Transients Without Scram (ATWS)

A number of plant transients can be affected by a failure of the scram system to function. For a pressurized water reactor, the most important transients affected include loss of normal feedwater, loss of electrical load, inadvertent control rod withdrawal, and loss of normal electrical power. In September 1973, we issued WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," establishing acceptance criteria for anticipated transients without scram. In conformance with the requirements of Appendix A to WASH-1270, Combustion Engineering submitted an evaluation of anticipated transients without scram in Topical Report CENPD-158, "Topical Report Anticipated Transients Without Scram." On December 9, 1975, we issued our "Status Report on Anticipated Transients Without Scram for Combustion Engineering Reactors." In response, CE issued Revision 1 to CENPD-158 in May 1976. A reevaluation of the potential risks from anticipated transients without scram (ATWS) has been published in NUREG-0460, Volumes 1 through 4. The current status of this issue is as follows:

(1) In March 1980 the 4th Volume of NUREG-0460 was issued by the NRC staff. The recommendations included design criteria for plants such as CESSAR and recommended rulemaking to establish such criteria.

(2) The NRC staff presented its recommendations on ATWS to the Commission, including the recommendation for rulemaking, in September 1980.

(3) After deliberation, the Commission will act on the matter. Whether it will agree to rulemaking is speculative at this time. If rulemaking is initiated by the Commission, we would expect that any rule adopted would include an implementation plan for all classes of plants.

All CESSAR plants would be required to provide plant modifications in conformance with ATWS criteria and schedular requirements provided in the rule or as adopted by the Commission. The following discussion presents the bases for operation of CESSAR plants prior to the adoption of a rule.

In NUREG-0460, Volume 3, we state: "The staff has maintained since 1973 (for example, see pages 69 and 70 of WASH-1270) and reaffirms today that the present likelihood of severe consequences arising from an ATWS event is acceptably small and presently there is no undue risk to the public from ATWS. This conclusion is based on engineering judgment in view of: (a) the estimated arrival rate of anticipated transients with potentially severe consequences in the event of scram failure; (b) the favorable operating experience with current scram systems; (c) the limited number of operating reactors." In view of these considerations and our expectation that the necessary plant modifications will be implemented in 1 to 4 years following a Commission decision on anticipated transients without scram, we have generally concluded that pressurized water plants can continue to operate because the risk from anticipated transients without scram events in this time period is acceptably small. As a prudent course, in order to further reduce the risk from anticipated transients without scram events during the interim period before completing the plant modification determined by the Commission to be necessary, we have required that the following steps be taken:

(1) Develop emergency procedures to train operators to recognize anticipated transients without scram events, including consideration of scram indicators, rod position indicators, flux monitors, pressurizer level and pressure indicator, and any other alarms annunciated in the control room with emphasis on alarms not processed through the electrical portion of the reactor scram system.

(2) Train operators to take actions in the event of an anticipated transients without scram, including consideration of manually scramming the reactor by using the manual scram button, prompt actuation of the auxiliary feedwater system to assure delivery to the full capacity of this system, and initiation of turbine trip. The operator should also be trained to initiate boration by actuation of the high pressure safety injection system to bring the facility to a safe shutdown condition.

The staff will require that all applicants referring CESSAR commit to the above requirements. We consider these procedural requirements an acceptable basis for interim operation of the facilities based on our understanding of the plant response to postulated anticipated transients without scram events.

15.3.10 Conclusions

CESSAR has presented results for various accidents which meet NRC acceptance criteria as detailed in Section 15 of the Standard Review Plan or on an alternate basis as previously described (Section 15.3.2). With the exceptions noted below, CESSAR has provided adequate protection systems to mitigate accidents in compliance with the applicable GDC relating to core coolability, control rod insertability, and primary system pressure boundary integrity.

The following issues have not yet been completed:

1. Reactor coolant pump shaft seizure - CE will provide the results of reanalysis of reactor coolant pump shaft seizure accident to meet staff acceptance criteria.

2. Steam line breaks - We require that CE provide parametrics justifying the input parameters utilized in the analyses.

3. Steam line breaks - We require that CE perform verification analyses that illustrate the large steam line breaks are more limiting than small steam line breaks.

3. Feedwater system pipe breaks - We require that CE provide confirmatory analyses which illustrate that small feedwater line breaks with offsite power available, combined with the limiting single failure, will not result in exceeding 110% of the primary system design pressure.

15.4 Radiological Consequences of Design Basis Accidents

CESSAR has analyzed certain postulated design basis accidents in order to demonstrate the adequacy of the design in the mitigation of possible offsite radiological consequences. The accidents include:

Steam Line Break Accidents,
Reactor Coolant Pump Locked Rotor,
Steam Generator Tube Rupture,
Rod Ejection Accident,
Small Line Break Accident, and
Fuel Handling Accident (in part).

The loss-of-coolant accident (LOCA) and liquid tank rupture accident radiological consequences are site-specific, and will be reviewed in the applications referencing CESSAR. The staff has reviewed the CESSAR analyses and has also performed its own analyses in accordance with the applicable Standard Review Plans and Regulatory Guides. In calculating the radiological consequences of the accidents, the staff has represented the effects of atmospheric dispersion by the values of X/Q presented below. The Exclusion Area Boundary (EAB) X/Q is that value not expected to be exceeded by any plant referencing CESSAR. (The 2-hour EAB X/Q of 2.5E-3 is different from the value of 2.0E-3 in CESSAR.) Low Population Zone Boundary (LPZ) X/Q's are not specified in CESSAR. The staff has, therefore, used the most conservative values for the plants that currently referrence CESSAR (Table 1.4-1). The limiting values were obtained from the Construction Permit SER's for these plants.

For those accidents addressed in CESSAR, we have established the following site-related interface requirements for CESSAR reference plants, on the basis of the analyses described later in this section:

(1) Primary coolant activity: 0.1 μCi/gm dose equivalent I-131, secondary coolant maximum equilibrium fission product concentration; 1.0 μCi/gm dose equivalent I-131, primary coolant maximum equilibrium fission product concentration.

(2) Steam generator tube leakage: 0.1 gpm primary to secondary.

(3) Containment leakage rate: 0.1%/day.

(4) Atmospheric dispersion factors (X/Q, sec/m^3) equal to or less than: 2.5E-3 for 0-2-hour Exclusion Area Boundary, 1.0E-4 for 0-8-hour Low Population Zone; 2.8E-5 for 1-4-day Low Population Zone; and 8.3E-6 for 4-30-day Low Population Zone.

These values may charge as the result of our review of additional analyses to be provided by CE.

15.4.1 <u>Main Steam Line Break</u>

Both the staff and CE have evaluated the radiological consequences of a postulated steam line break accident occuring outside containment and upstream of the main steam isolation valve. The staff's evaluation has been carried out in accordance with the procedures and using the criteria specified in Standard Review Plan 15.1.5, Appendix A. Although the contents of the secondary side of the affected steam generator would be vented initially to the atmosphere as an elevated release, we have conservatively assumed that the entire release throughout the accident is at ground level. During the course of the accident, the staff assumed the shell side of the affected steam generator stayed dry because the auxiliary feedwater flow to this steam generator would be blocked off under conditions of this accident. Due to the dry-out condition in the affected steam generator, all iodine transported to the secondary side by primary-to-secondary leakage (Technical Specification limit) was assumed to be available for release to the atmosphere with no reduction due to holdup or attenuation.

The staff has evaluated three potential cases for this accident:

(1) an iodine spike is generated as a result of the accident, and the iodine release rate from the fuel to the coolant is increased by a factor of 500. Prior to the accident, the plant was assumed to be operating at a primary coolant activity level of 1 μCi/gm, dose equivalent I-131. This value is different from that proposed by CE for CESSAR (4.6 μCi/gm), but it is a value identified in all PWR Standard Technical Specifications including CE's;

(2) a preaccident iodine spike has occurred which raised the primary coolant iodine concentration to 60 μCi/gm, dose equivalent I-131; and

(3) the most reactive control rod fails to insert.

In absence of secondary side releases for this accident, the staff is unable to evaluate the radiological consequences of this third case and this will be reported in a revision to this report when CE supplies this information. It should be noted that the conclusions reached here are based only on the first two cases.

The staff's calculated doses for these two cases are presented in Table 15.4-1. The assumptions used in the staff's analysis are presented in Table 15.4-2. For the case of a time-dependent iodine spike, the staff's analysis has shown that the calculated thyroid doses exceed the criterion (small fraction of the 10 CFR Part 100 exposure guidelines) specified in the acceptance criteria of SRP Section 15.1.5, Appendix A. This criterion would be met if the limiting Technical Specification value for primary-to-secondary leakage were reduced from the proposed (and standard) technical specification value of 1 gpm to 0.3 gpm.

For the case of preaccident iodine spike, the calculated thyroid dose is within the 10 CFR Part 100 guidelines based on the interface requirements set forth at the beginning of this section, and the CESSAR design basis is, therefore, acceptable.

The third case has not yet been resolved.

The staff will also find applications that references CESSAR acceptable with respect to the main steam line break accident which:

(1) has site meteorology which results in X/Q's less than or equal to the following values:

    0-2 hour Exclusion Area Boundary X/Q of 2.5 E-3 sec/cubic meter and
    0-8 hour Low Populatio Zone Boundary X/Q of 1.0 E-4 sec/cubic meter.

(2) has proposed Technical Specifiction limits equal to or less than those specified below:

    primary-to-secondary leakage of 0.3 gpm; equilibrium primary coolant activities of 1 μCi/gm dose equivalent I-131 and 100/Ē μCi/gm gross activity and 60 μCi/gm spiking limit for dose equivalent I-131; and secondary coolant activity limit of 0.1 μCi/gm dose equivalent I-131.

15.4.2 Reactor Coolant Pump Locked Rotor Shaft Seizure

CE has indicated that a reactor coolant pump locked rotor accident could lead to 17.6% fuel clad failure. Such failure results in the release of 10% of the radioiodine and noble gas inventory of the affected fuel rods. The fission products are released to the environment through steam generator tube leakage to the secondary system. Because of the assumption of loss of offsite power, the main turbine condensers are assumed to be unavailable during the recovery from this accident, and it was assumed that the steam is dumped to the environment by a combination of the operation of the automatic safety valves and manual atmospheric relief valves. CE's and the staff's analyses also assumed that the Technical Specification primary-to-secondary leakage is occurring to only one

TABLE 15.4-1

RADIOLOGICAL CONSEQUENCES OF LARGE STEAM LINE BREAK ACCIDENT

Case 1: Iodine Spike Occurs at Time of Accident

	Thyroid	Whole Body
EB Dose, rem	30	<1
LPZ Dose, rem	1.5	<1

Case 2: Pre-accident Iodine Spike of 60 μCi/gm

	Thyroid	Whole Body
EB Dose, rem	51.6	<1
LPZ Dose, Rem	2.1	<1

TABLE 15.4-2

ASSUMPTIONS USED FOR THE STEAM LINE BREAK ACCIDENT

PARAMETER	ASSUMED VALUE
Initial core power level, Mwt	3876
Percent of fuel failure	0
Primary-to-secondary leakage, gpm	1
Preaccident iodine spike, μCi/gm	60
Primary coolant concentration, μCi/gm	1
Secondary side concentration, μCi/gm	0.1
Affected steam generator partition factor	1
Unaffected steam generator partition factor	1

generator for the accident duration and that this generator was initially at the secondary coolant activity limit of 0.1 μCi/gm, I-131 dose equivalent.

In the analysis of this accident, the staff has used a partition factor of 100 between water and vapor phases in the steam generator. This value is justified because leakage between primary and secondary systems is very low (1 gpm) and typically occurs at the tube sheet in U-bend steam generators. In addition, corrosion inhibitors added to the secondary side would result in considerably more iodine retention by the secondary side water.

CE's and the staff's analyses indicate that the radiological consequences for this accident would be about 100 rem (thyroid) if the CE Standard Technical Specification for primary-to-secondary leakage is assumed for the course of this accident. Both CE's and the staff's calculated radiological consequences exceed the acceptance criteria of SRP Section 15.3.3, i.e., small fraction of (less than or equal to) the 10 CFR Part 100 guideline values. In order to meet the acceptance criteria, the amount of primary-to-secondary leakage in the steam generators must be reduced from the CE Standard Technical Specification limit of 1 gpm to 0.1 gpm. Using this restriction on the primary-to-secondary leakage, the staff finds the CESSAR design acceptable with respect to this accident.

The staff will find an application that references CESSAR acceptable with respect to the locked rotor accident which:

(1) has site meteorology which results in X/Q's less than or equal to those specified in Section 15.4.1.

(2) has proposed a Technical Specification limit on primary-to-leakage less than or equal to 0.1 gpm.

In a letter dated October 29, 1981. CE indicated that they are reanalyzing this accident to demonstrate conformance with the SRP acceptance criterion. We will report on the resolution of the issue in a revision to this report.

15.4.3 Rod Ejection Accident

For this accident, a mechanical failure of the control rod drive mechanism is postulated. As a result of the failure, primary coolant would leak to the containment and the control rod and drive shaft would be moved to the fully withdrawn position. The consequence of this mechanical failure is a rapid positive reactivity insertion and primary system depressurization. This leads to an adverse core power distribution and localized fuel damage. CE has calculated that 9.8% of all fuel rods will experience cladding failure. The staff has used this estimate. Fuel temperatures, however, are not expected to cause fuel melting.

The release of radioisotopes from this accident are calculated for two pathways. One path is through containment leakage which results from release of primary coolant to the containmetn through the ruptured drive mechanism. The second path is through the release of contaminated steam from the secondary system. The contamination of the secondary system is assumed to be due to the leakage of primary coolant to the secondary side of the steam generators by tube leakage. The primary-to-secondry leakage is assumed to be at the CE Standard Technical Specification limit of 1 gpm. CE has also proposed this limit in the technical specifications for CESSAR. The primary-to-secondary leak is assumed to subside after the pressure between the two systems equalizes which is about 925 seconds after the accident.

The staff's analysis of this accident used a partition factor of 100 between water and vapor phases in the steam generator as discussed in the previous section. The radiological consequences were evaluated using the guidance of Standard Review Plan Section 15.4.8 (Appendix A) and the recommendations of Regulatory Guide 1.77. The assumptions used for the two release paths are given in Table 15.4-3. The calculated radiological consequences are presented in Table 15.4-4 and are well within the guideline values of 10 CFR Part 100. Therefore, the acceptance criteria of the SRP are met and the staff concludes that the CESSAR design for limiting the consequences of this postulated accident and the CESSAR Technical Specifications (for primary-to-secondary leakage) coupled with the CESSAR limiting X/Q are acceptable.

The staff will also find an application that references CESSAR acceptable with respect to the control rod ejection accident which:

(1) has site meteorology which results in X/Q's better than or equal to those specified in Section 15.4.1.

(2) has proposed a Technical Specification limit on the primary-to-secondary leakage less than or equal to 1 gpm.

15.4.4 Failure of Small Lines Carrying Primary Coolant Outside Containment

The most severe rupture of the small lines carrying primary coolant outside containment during normal operation would be the double ended break of the chemical volume control system (CVCS) letdown line just outside containment. The applicant has estimated that the CVCS letdown line will be isolated in 10 minutes following the break and that 30,766 pounds of primary coolant will be released.

The assumption of operator action within 10 minutes following the accident is based on the proposed ANS 58.8, ANSI N660, Rev. 2, 1981, "Time Response Design Criteria for Safety-Related Operator Actions." Table 15.6.2-1 of CESSAR lists a number of alarms associated with this accident, including three alarms (RHX exit high temperature alarm, letdown line low flow and low pressure alarms, and the low flow alarms in the Process Radiation Monitor and the Boronometer) which alert the operator immediately after event initiation.

In the staff's analysis, the primary coolant's maximum equilibrium fission product concentration was taken to be at 1 μCi/gm*, dose equivalent I-131, followed by a 500-fold increase in the release rate of iodine from the fuel during the accident.

The iodine in the released coolant which flashed to steam was assumed to be released directly to the environment without taking credit for filtration or plateout.

*The primary coolant activity level of 1 μCi/gm is different from the value proposed by CE for CESSAR System 80 (4.6 μCi/gm) but this value is identified in all PWR Standard Technical Specifications, including CE's.

TABLE 15.4-3 Assumptions Used for the Control Rod Ejection Accident

Parameter	Assumed Value
Initial core power level, Mwt	3876
Percent of fuel failure	9.8
Primary-to-secondary leakage, gpm	1
Peaking factor	1.5
Primary system volume, lbm	5.34 E5
Containment leak rate, %/day	0.1

TABLE 15.4.4 Radiological Consequences of a Control Rod Ejection Accident

	Radiological Consequences (REM)			
	EAB		LPZ	
Leakage Path	Thyroid	Whole Body	Thyroid	Whole Body
Containment	110	2.4	62	0.4
Secondary Side	96	3.4	92	0.8

The calculated thyroid and whole body doses, using the primary coolant activity of 1 μCi/gm and enveloping X/Q of 2.5 E-3 sec/cubic meter, are found to be 23 rems and $<$ 1 rem, respectively. These doses meet the acceptance criteria of Standard Review Plan Section 15.6.2. The use of the 4.6 μCi/gm, currently proposed by CESSAR, in the calculation will, however, result in doses which exceed the acceptance criteria. The staff will, therefore, require that reference plant applicants use the CE Standard Technical Specification limit on the primary coolant activity of 1 μCi/gm, dose equivalent I-131.

15.4.5 Steam Generator Tube Rupture Accident

CE has not yet completed their analysis of this accident.

On the basis of our experience with the evaluations of steam generator tube rupture accidents for PWR plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary and secondary coolant system radioactivity concentrations so that potential offsite doses meet the acceptance criteria of Standard Review Plan Section 15.6.3. These limits can be included in the Technical Specifications for those plants referencing CESSAR. We will report on the resolution of this issue in a revision to this report.

15.4.6 Fuel Handling Accident

Two parameters proposed by CE for analysis of this accident depart from those used by the staff in its review of such accidents. CE proposes that the fuel handling accident analysis use a peaking factor be 1.55 instead of the 1.65 value set forth in Regulatory Guide 1.25. In addition, CE proposes that such analyses use 60 fuel rods versus the full assembly (236 fuel rods) utilized by SRP Section 15.7.4. The staff's analysis of this accident would utilize the parameters outlined in Regulatory Guide 1.25 and the guidance of SRP Section 15.7.4 (1.65 peaking factor and 236 rods damaged).

Until CE either provides acceptable justification for the assumptions used in their analysis or amends CESSAR to conform to the guidelines of Regulatory Guide 1.25, with regard to the parameters described above, we will evaluate the fuel handling accident separately on each application referencing CESSAR.

16 TECHNICAL SPECIFICATIONS

The Technical Specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the Commission. The approved Technical Specifications, for the CESSAR scope, will be made a part of the operating license for facilities referencing CESSAR. Included will be sections covering safety limits, limiting safety systems settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

CE has proposed that the Technical Specifications given in Chapter 16 of the CESSAR Final Safety Analysis Report be used. The NRC staff is currently working with CE to update the proposed Technical Specifications to include the results of the staff's review of the systems and equipment within the CESSAR scope. The staff's final conclusions will be reported in a revision to this report.

17 QUALITY ASSURANCE

17.1 General

The description of the quality assurance program for the design, procurement, and fabrication of CESSAR is contained in Chapter 17 of the FSAR which references CE topical report CENPD-210A, Revision 3, "Quality Assurance Manual for NSSS." We have evaluated the quality assurance program description because it differs from the description provided in the CESSAR Preliminary Safety Analysis Report. Our evaluation of this later quality assurance program is based on a review of the information provided in the FSAR and discussions with representatives from CE. We assessed the CE quality assurance program to determine if it complies with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," the applicable quality assurance related regulatory guides listed in Section 17 of the Standard Review Plan.

17.2 Organization

The organization responsible for the design, procurement, and fabrication of the nuclear steam supply system is the Power System Group, shown in Figure 17.1-1. This group consists of three organizations: General Services, Nuclear Power Systems, and Power Systems Services, each directed by a Vice-President who reports to the President of the Power Systems Group.

The Vice-President, General Services, through the Director, Group Quality Assurance defines the quality assurance program of the group and ensures compliance with the program by auditing throughout the group. The Director, Quality Assurance, determines that the mandatory quality assurance program requirements are imposed on management by means of the Quality Assurance Policy Manual and the Power Systems Group Nuclear Quality Assurance Manual.

CE management is required by the Group Nuclear Quality Assurance Manual to develop systems and procedures to implement the mandatory quality assurance policy. The Director, Group Quality Assurance, audits to verify that this requirement is met. The organizations directly responsible for implementation of the quality assurance program include Group Quality Assurance. Engineering Quality Assurance, the manufacturing quality assurance groups, and the quality assurance group in Construction Services. The Director, Group Quality Assurance, in addition to developing the quality assurance program, is responsbile for supplier control. This includes control of the CE manufacturing organizations. The supplier control program includes (1) evaluation and approval of quality assurance programs, (2) review and approval of procurement orders, (3) surveillance audit, and (4) review and approval of procedures, records, and certifications.

The right to stop work is delegated from the President to all levels of quality assurance management. Disputes between personnel in quality assurance and personnel in engineering, purchasing, manufacturing, or supplier organizations are

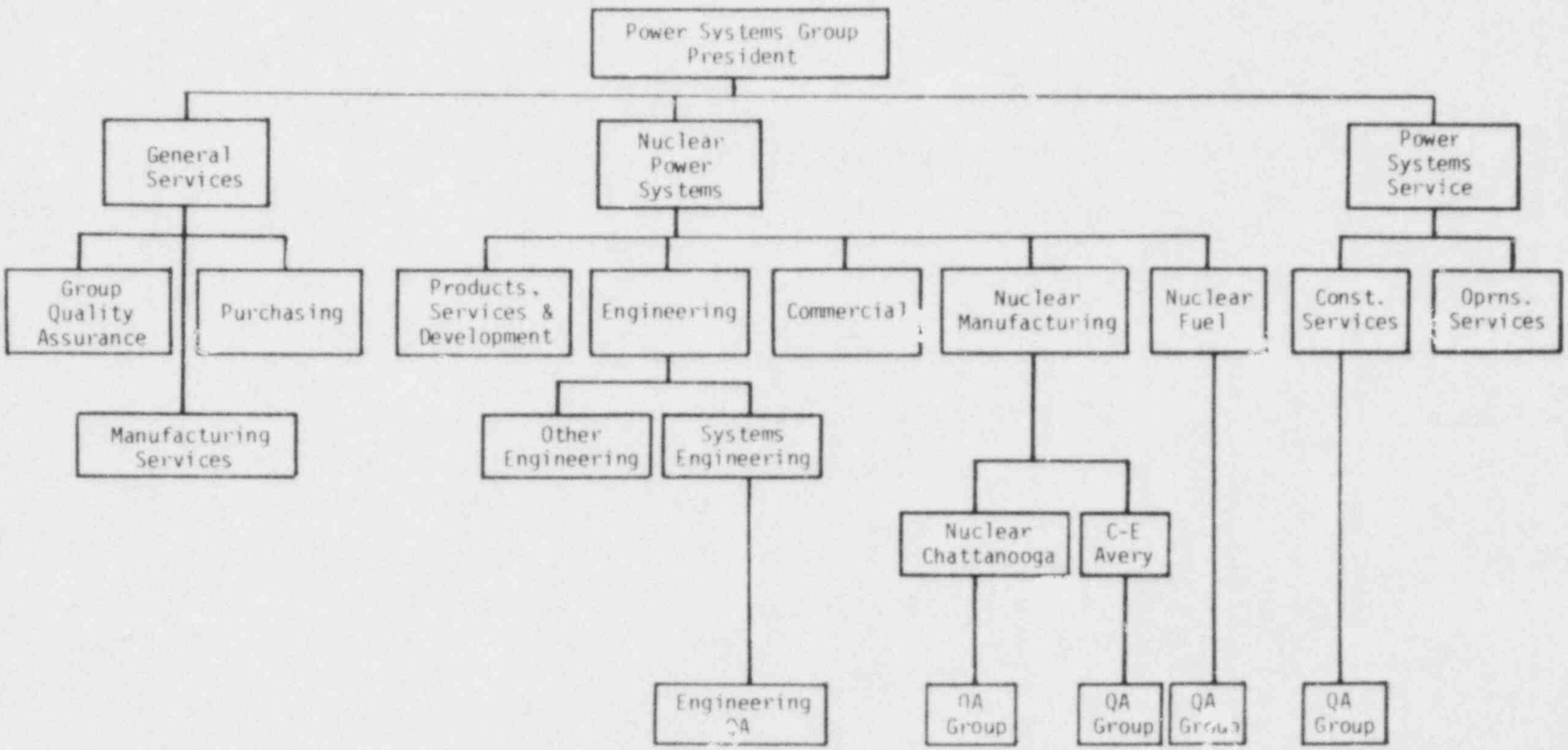

FIGURE 17.1 - Power Systems Group

settled at the Director or Vice-President level when not resolved at lower levels.

The CE quality assurance organizations have the authority and freedom to identify quality problems; to initiate, recommond, or provide solutions; to verify implementation of solutions; and to control further processing, delivery, or installation of a nonconforming item until proper disposition of the deficiency or unsatisfactory condtion has been approved. We conclude that the CE quality assurance organizations comply with Appendix B to 10 CFR 50 and are, therefore, acceptable.

17.3 Quality Assurance Program

CE has provided a cross-reference which identifies the procedures and manuals which implement each of the criteria of 10 CFR Part 50, Appendix B. The quality assurance program commits CE to meet the requirements of Appendix B to 10 CFR Part 50. Also, CE has committed to comply with the regulatory position of applicable NRC QA regulatory guides and ANSI standards listed in Table 17.1-1, with some exceptions, as described in the Topical Report CENPD-210 Revision 3, which the staff has found acceptable. We find, with these commitments and our review of the CE quality assurance polices and quality assurance program description, that CE has defined an acceptable quality assurance program.

The quality assurance program applies to all safety-related items and services engineered, procured, and manufactured by CE, including nuclear fuel assemblies. Highlights of the quality assurance program are described below.

Procedures require formal training and indoctrination of personnel performing activities affecting quality to assure they are suitably trained and their proficiency is maintained.

The quality assurance program provides a system for design control. The system is documented and controlled by procedures and instructions. These procedures and instructions describe the responsibilities and interfaces of each organizational unit which has an assigned responsibility. Distribution lists and master lists of project drawings and specifications are maintained to assure timely and accurate access to latest applicable documents. Procedures are established for verifying designs.

CE has established and documented measures for the preparation, review, approval, and control of procurement documents. These measures provide assurance that the procurement documents include or reference regulatory requirements, design bases, and quality requirements.

Group Quality Assurance reviews and approves purchase specifications prior to issuance. Reviews of procurement documents by qualified engineering and quality assurance personnel provide assurance that quality requirements are complete and correctly stated. The reviews also assure that the quality requirements can be controlled by the supplier/manufacturer and verified by Group Quality Assurance personnel.

CE requires that its suppliers/manufacturers identify and control materials, and Group Quality Assurance inspects the marking of items prior to shipment.

TABLE 17.1-1 Regulatory Guidance Applicable to Quality Assurance Program

1. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)" (6/72).

2. Regulatory Guide 1.30, "Quality Assurance Requirements fo the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment" (8/11/72).

3. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" (3/16/73).

4. Regulatory Guide 1.38 - Revision 2, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants" (5/77).

5. Regulatory Guide 1.39 - Revision 2, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants" (9/77).

6. Regulatory Guide 1.58 - Revision 1, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel" (9/80).

7. Regulatory Guide 1.64 - Revision 2, "Quality Assurance Requirements for the Design of Nuclear Power Plants" (6/76).

8. Regulatory Guide 1.74, "Quality Assurance Terms and Definitions" (2/74).

9. Regulatory Guide 1.88 - Revision 2, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records" (10/76).

10. Regulatory Guide 1.94 - Revision 1, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants" (4/76).

11. Regulatory Guide 1.116 - Revison 0-R, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems" (5/77).

12. Regulatory Guide 1.146, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants" (8/80).

13. ANSI N45.2.12 - Draft 3, Revision 4, "Auditing of Quality Assurance Programs for Nuclear Power Plants" (2/74).

13. ANSI N45.2.13 - Draft 2, Revision 4, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants" (4/74).

Material identification and control is assured by requiring a written procedure which is reviewed by Group Quality Assurance.

CE requires that in-process and final inspections be performed in accordance with procedures submitted to and found acceptable by CE. Procedures require that inspection personnel be qualified and that records of qualification be maintained. These procedures require that inspection personnel be organizationally independent from personnel who perform the work being inspected.

CE suppliers/manufacturers must maintain a system providing for identification, documentation, and control of nonconforming items to prevent inadvertent use. Group Quality Assurance reviews and approves nonconformance actions. Engineering evaluates and dispositions nonconformances, and Group Quality Assurance reviews these actions. Group Quality Assurance also verifies proper corrective action. CE provides nonconformance reports which are dispositioned "use as is" and "repair" to the utility with the affected item.

CE executes a comprehensive system of planned and documented audits to verify product quality and compliance with the quality assurance program. The audits, with preestablished check lists, assure compliance with all aspects of 10 CFR 50, Appendix B, including the quality-related aspects of design, procurement, manufacture, storage, shipment, and reactor site activities. The CE quality assurance program requires that suppliers/manufacturers also audit their operations and their subvendor's operations to verify conformance with quality requirements. The audits include quality-related practices, procedures, instructions, and conformance with the quality assurance program. Group Quality Assurance conducts audits of the CE suppliers/manufacturers and selected subvendors. Written reports are forwarded to management of the area audited and to CE management. Follow-up audits assure corrective action.

In this review, the staff has evaluated the CE quality assurance program covering safety-related structures, systems, and components for compliance with regulations and applicable guidance provided by the NRC. Based on the review, the staff concludes that CE has described a quality assurance program that contains the necessary quality assurance provisions, requirements, and controls for compliance with Appendix B to 10 CFR Part 50 and applicable guides and standards and is acceptable for the CESSAR nuclear steam supply system.

17.4 Conclusion

Our review of the CE quality assurance program description for CESSAR has established and verified that all applicable requirements of Appendix B to 10 CFR Part 50 are included in the quality assurance program. Further, this review established that the quality assurance organizations are structured such that they can effectively carry out their responsibilities related to quality without undue influence from other groups. Our determination of acceptability included a review of the list of items to which the quality assurance program applies (CESSAR Table 3.2-1). The list of items was reviewed by each technical review branch in the NRC to assure that the safety-related items within their scope of review are under the quality assurance program controls. CE revised CESSAR Table 3.2-1 in a letter dated November 3, 1981 to incorporate staff comments. In addition, this list has been expanded to include safety-related items reflected in the NUREG-0737 requirements (see Section 22).

Based on our detailed review and evaluation of the quality assurance program description contained in the FSAR for CESSAR, we conclude that:

1. The quality assurance organizations within CE are provided sufficient independence from cost and schedule (when opposed to safety considerations), authority to effectively carry out the quality assurance programs, and access to management at a level necessary to perform their quality assurance functions.

2. The quality assurance program describes adequate quality assurance requirements and controls which, when properly implemented, comply with the criteria of Appendix B to 10 CFR Part 50.

Accordingly, we concude that the CESSAR quality assurance program is in compliance with the applicable NRC regulations and is, therefo , acceptable.

18 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The CESSAR appliction for an FDA is being reviewed by the Advisory Committee on Reactor Safeguards. The NRC staff will issue a revision to this Safety Evaluation Report after the Committee report to the Commission is available. The revision will append a copy of the Committee's report, will address comments made by the Committee, and will describe steps, taken by the NRC staff to resolve any issues raised as a result of the Committee's review.

19 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are United States citizens. The applicant is not owned, dominated, or controlled by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant agreed to safeguard any such data which might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes will be involved. For these reasons and in the absence of any information to the contrary, the staff finds that the activities to be performed will not be inimical to the common defense and security.

20 FINANCIAL QUALIFICATIONS

The financial qualifications of the utility will be addressed in the SER for the applications that reference CESSAR.

21 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

The indemnity requirements of 10 CFR Part 140 will be addressed in the SER for the applications that reference CESSAR.

22 TMI-2 REQUIREMENTS

22.1 Introduction

The accident at Three Mile Island Unit 2 (TMI-2) resulted in requirements which were developed from the recommendations of several groups established to investigate the accident. These groups include the Congress, the General Accounting Office, the President's Commission on the Accident at Three Mile Island, the NRC Special Inquiry Group, the NRC Advisory Committee on Reactor Safeguards, the Lessons Learned Task Force and the Bulletins and Orders Task Force of the NRC Office of Nuclear Reactor Regulation, the Special Review Group of the NRC Office of Inspection and Enforcement, the NRC Staff Siting Task Force and Emergency Preparedness Task Force, and the NRC Offices of Standards Development and Nuclear Regulatory Research. The report NUREG-0660, entitled "NRC Action Plan Developed as a Result of the TMI-2 Accident" (referred to as Action Plan), was developed to provide a comprehensive and integrated plan for the actions now judged necessary by NRC to correct or improve the regulation and operation of nuclear facilities. The Action Plan was based on the experience from the TMI-2 accident and the recommendations of the investigating groups.

With the development of the Action Plan (NUREG-0660), NRC has transformed the recommendations of the investigating groups into discrete scheduled tasks that specify changes in its regulatory requirements, organization, or procedures. Some actions to improve the safety of operating plants were judged to be necessary before an action plan could be developed, although they were subsequently included in the Action Plan. Such actions came from the Bulletins and Orders issued by the Commission immediately after the accident, the first report of the Lessons Learned Task Force, and the recommendations of the Emergency Preparedness Task Force. Before these immediate actions were applied to operating plants, they were approved by the Commission.

NRC has identified a discrete set of licensing requirements related to TMI-2 in the Action Plan for plants that are scheduled to receive an operating license in the near future. The report NUREG-0737, entitled "Clarification of TMI Action Plan Requirements," was issued in November 1980. This report identifies the specific items from NUREG-0660 that have been approved by the Commission for implementation at nuclear power plants. It also includes additional information about schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions. This section summarizes the NRC staff review of CESSAR against the criteria of NUREG-0660, as clarified by NUREG-0737.

CE addressed the NUREG-0737 guidelines in Appendix B to the CESSAR FSAR. The majority of guidelines in NUREG-0737 relate to site-specific and utility organizational matters, which are outside the scope of CESSAR and will be addressed on applications referencing CESSAR. Those requirements that are within the CESSAR scope are as follows:

Action Plan Item	Title
II.B.1	Reactor Coolant System Vents
II.D 1	Relief and Safety Valve Test Requirements
II.E.1.2	Auxiliary Feedwater System Initiation and Flow
II.E.4.2	Containment Isolation Dependability
II.F.2	Instrumentation for Detection of Inadequate Core Cooling
II.K.1.5	IE Bulletins - Review ESF Valves
II.K.2.13	Thermal Mechanical Report
II.K.2.17	Voiding in RCS
II.K.3.5	Auto Trip of RCP's
II.K.3.25	Power on Pump Seals
II.K.3.30	Small Break LOCA Methods
II.K.3.31	Plant Specific Analysis
III.D.1.1	Primary Coolant Outside Containment

22.2 Evaluation

The guidelines for each of the issues identified above are described in NUREG-0737. Our evaluation of the CESSAR design relative to each of these issues is summarized below.

II.B.1 Reactor Coolant System Vents

CESSAR provides a connection on the reactor vessel head for venting. Specific valve and piping designs from this point on are provided by the reference plants. A similar connection point is provided for the pressurizer. Both systems are designed to reactor coolant system pressure boundary quality standards.

We conclude that this part of the vent design is acceptable. The bulk of the vent system design will be provided by the applications referencing CESSAR, including procedures for venting other parts of the reactor coolant system.

II.D.1 Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves

CE, through its technical support to the PWR Owners' Group, participates in the EPRI/NSAC program to conduct performance testing of PWR relief and safety valves and associated piping and supports. CE has referenced the proposed EPRI program ("Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems," dated December 13, 1979) for the performance testing of these valves. Additionally, by letter dated December 15, 1981, the Owners' Group has responded to the clarification requirements of NUREG-0737.

A description of the EPRI/NSAC program was provided to the NRC in December 1979 and an updated revision to the program was submitted in July 1980. The staff has reviewed these program descriptions and is generally in agreement that the NUREG-0737 technical guidelines for relief and safety valves and associated piping and supports can be met.

Inasmuch as the resolution of this issue is dependent upon plant-specific piping arrangements which are outside the CESSAR scope, we will address this issue on each application referencing CESSAR.

II.E.1.2 Auxiliary Feedwater System Automatic Initiation and Flow Indication

The emergency feedwater automatic initiation is part of the ESFAS (see Section 7.3) and conforms to the requirements for protection systems in accordance with IEEE Standard 279. Therefore, we conclude that this system conforms to the Action Plan guidelines. Other guidelines of this issue are outside the CESSAR scope and will be addressed on each application referencing CESSAR.

II.E.4.2 Containment Isolation Dependability

The staff has reviewed the actuation and containment isolation capabilities of the ESFAS, as described in Sections 6.2.4 and 7.3 of this report. Based on that review we conclude that the applicable portions of the CESSAR design conform to the Action Plan guidelines.

II.F.2 Instrumentation for Detection of Inadequate Core Cooling

The review of this issue, as it pertains to the CESSAR design, is not complete. We will report on the resolution of this issue in a revision to this report.

II.K.1.5 IE Bulletin on Measures to Mitigate Small-Break LOCAs and Loss of Feedwater Accidents

Systems from service (and restoring to service) to assure operability status is known.

In response to the above issue (NUREG-0737, Enclosure 2), CE has stated in Appendix B to the FSAR, that the valves of the engineered safety features (ESF) systems are designed and tested to ensure proper operation in the event of an accident. This is accomplished in the following ways:

1. The valves of the ESF systems are interlocked to automatically provide the sequence of operations required after an actuation of the ESF.

2. Actuator-operated valves are provided with key-operated control switches, where considered necessary, to prevent unintentional misalignment of safety injection flow paths during power operation.

3. All valves that are not required to operate on initiation of safety injection or recirculation, in the injection flow path, are locked in the post-accident position. Administrative controls ensure that the valves are locked in the correct position.

4. Periodic tests and inspections are performed by the applicant using CESSAR to verify proper operation of each active component of the safety injection system. This includes valves.

We find the CESSAR response meets the technical guidelines of NUREG-0737 and is, therefore, acceptable.

II.K.2.13 Thermal Mechanical Report--Effect of High-Pressure Injection Vessel Integrity for Small-Break Loss-of-Coolant Accident With No Auxiliary Feedwater

A program which completely addresses the NRC requirements of detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater will be completed, documented, and submitted to NRC by January 1, 1982. This program will consist of analyses for generic CE PWR plant groupings. If required, the generic analyses will be supplemented by plant-specific analysis.

Based on the above, we conclude that the CESSAR commitment meets the schedular guidelines of this item and is acceptable.

II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients

In response to the above issue, Amendment 2 to the FSAR has stated that CE is participating in the CE Owners Group evaluation of the generic applicability of these guidelines. CE has committed to provide the results of this analysis by January 1, 1982. We conclude that this commitment meets the implementation schedule of this item in NUREG-0737 and, therefore, is acceptable. We will condition the reference plant license, if necessary, to require compliance with this implementation schedule.

II.K.3.5 Automatic Trip of Reactor Coolant Pumps During LOCA

In response to the above issue, CE has stated, in Amendment 2 to the FSAR, that, through their participation in the CE Owners Group, they are continuing to study the effects of RCPs on small-break LOCAs and the possible need for automatic RCP trip.

A report (CEN-115) has been provided to the staff by the CE Owners Group which contains the results of a generic study of the influence of RCPs on small-break LOCA transients. Following model verification through comparison with integral test data, the need for an automatic RCP trip will be reassessed. Based on the above, we conclude that CESSAR has met the guidelines of this item.

II.K.3.25 Effects of Loss of AC Power on Pump Seals

In response to the above issue, CE has stated, in its letter dated October 9, 1981, that the reactor coolant pump seals are cooled by redundant systems, i.e., seal injection water and component cooling water. In the event of loss of ac power, seal injection water can be restored by manually furnishing essential power to the charging pumps. The interface requirement to furnish essential power to the charging pumps is specified in Section 9.3.4.6 of CESSAR.

Essential power should be restored to the charging pumps within 20 minutes. It is a design requirement that the pump seals be capable of withstanding a loss of component cooling water and seal injection water without damage in the event of loss of offsite ac power. Once seal injection has been restored the pump seals are capable of withstanding hot standby operation for in excess of 2 hours. This capability has been demonstrated by test on a production System 80 reactor coolant pump. CE has also provided test data that indicates the maximum pump seal temperatures are within acceptable limits without component cooling water supply to the pump seal heat exchangers. However, CE assumed that the seal injection flow is available at the time of loss of offsite ac power, and neglects the effects of the time delay for the manual action in restoring seal injection flow to the RCPs. We require that CE provide additional information which will demonstrate that the RCP seals can withstand 2 hours without seal failure following loss of offsite ac power. Otherwise, we will require that CE incorporate, in the CESSAR design, an interface requirement which requires essential power buses to backup the component cooling water system operation.

We will report our resolution in a revision of this issue of this report.

II.K.3.30 Revised Small-Break LOCA Methods to Show Compliance with 10 CFR Part 50, Appendix K

In response to the above requirements, CE has committed, in a letter dated October 9, 1981, to submit the final report to justify the adequacy of the present small-break LOCA model by January 1, 1982.

We conclude that this commitment meets the implementation schedule of NUREG-0737 and, therefore, is acceptable.

II.K.3.31 Plant-Specific Calculations to Show Compliance With 10 CFR 5.46

In response to the above requirements, the CE has committed, in a letter dated October 9, 1981 to submit within 1 year after staff approval of the SBLOCA models, revised SBLOCA ECCS analyses for CESSAR.

We conclude that this commitment meets the implementation schedule of NUREG-0737 and is, therefore, acceptable.

III.D.1.1 Integrity of System Outside Containment Likely to Contain Radioactive Material for Pressurized Water Reactors and Boiling Water Reactors

We have reviewed the CE submittals relating to TMI Action Plan Item III.D.1.1 of NUREG-0737. In these submittals, CE has described the measures that have been incorporated for controlling leakage from systems outside the containment that would or could contain highly radioactive fluids during serious transient or accident conditions. The systems that come within the scope of CESSAR final design in this regard are the chemical and volume control system, safety injection and shutdown cooling systems, and the containment spray system. These measures also include special provisions for the automatic isolation of the letdown flow upon safety injection actuation signal (SIAS) and the automatic containment isolation of reactor coolant pump controlled bleed-off upon containment isolation actuation signal (CIAS). The latter provision will result in redirection of reactor coolant pump controlled bleed-off to the reactor drain tank within the containment structure. These provisions are included as interface requirements for the reference plants.

Based on our review of the measures that have been incorporated to control leaks from systems that come within the scope of the CESSAR design, we have concluded that these measures conform to the Action Plan guidelines.

23 CONCLUSIONS

Based on the staff evaluation of the CESSAR final design as set forth in the preceding sections of this report, it is the staff's position that, subject to favorable resolution of the outstanding matters described herein, the staff will be able to conclude that:

(1) The application for a Final Design Approval (FDA) filed by Combustion Engineering, Inc. on October 27, 1978, as amended, complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1; and

(2) The facilities referencing CESSAR, subject to the approval of the balance-of-plant design, can conform with the provisions of the Act and the rules and regulations of the Commission; and

(3) There is reasonable assurance that (a) the activities authorized by licenses or permits referencing CESSAR, subject to the approval of the balance-of-plant design, can be conducted without endangering the health and safety of the public, and (b) such referencing will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter 1; and

(4) The issuance of the Final Design Approval will not be inimical to the common defense and security or to the health and safety of the public.

Before a Final Design Approval will be issued for the CESSAR design, those outstanding issues, described herein, must be resolved. Subsequently, applications submitted under 10 CFR 50.22, with a compatible balance-of-plant design, may reference the CESSAR FDA, provided that they (1) identify any deviations from the CESSAR design, as described in the CESSAR FSAR and herein; (2) demonstrate conformance with the interface requirements, as described in the CESSAR FSAR and herein; and (3) otherwise conform to the Act and the rules and regulations of the Commission.

APPENDIX A

CHRONOLOGY OF CESSAR REVIEW

October 27, 1978	Letter from CE tendering the CESSAR FDA application with FSAR for acceptance review.
March 23, 1979	Letter to CE identifying additional information required to docket CESSAR.
December 21, 1979	Letter from CE submitting revised FSAR for docketing.
December 18, 1980	Letter from CE submitting balance of FSARs for staff review.
February 17, 1981	Letter to CE requesting a comparison between the CESSAR and San Onofre designs.
February 24, 1981	Letter from CE submitting Amendment No. 1 to the FSAR.
March 1, 1981	Letter from CE describing plans for the instrumentation and control systems Independent Design Review (IDR).
March 3, 1981	Letter from CE describing design details for the confirmatory CESSAR piping analysis.
March 5, 1981	Meeting in Bethesda, MD, with CE and Arizona Public Service to discuss agendas for the instrumentation and control systems IDR.
March 25, 1981	Letter from CE submitting Amendment No. 2 to the FSAR.
March 30, 1981	Letter to CE discussing the instrumentation and control systems IDR schedule.
April 16, 1981	Letter to CE describing the CESSAR environmental qualification review.
May 5, 1981	Letter to CE transmitting the preliminary evaluation of the CESSAR mechanical design.
May 7, 1981	Meeting in Bethesda, MD, with CE management to discuss CESSAR review schedule.
May 18, 1981	Meeting in Bethesda, MD, to discuss issues related to the CESSAR environmental qualification program.
May 22, 1981	Letter to CE requesting additional information for the CESSAR review.

May 26, 1981	Letter to CE addressing concerns raised in May 7, 1981 meeting.
May 29, 1981	Letter from CE proposing an agenda for meeting with the Auxiliary Systems Branch.
June 2-4, 1981	Meeting in Windsor, CT - CESSAR instrumentation and control systems Independent Design Review.
June 9, 1981	Meeting in Bethesda, MD, to discuss the CESSAR auxiliary systems.
June 10, 1981	Letter to CE requesting FSAR Appendix A be updated to reflect latest revisions to Regulatory Guides.
June 11, 1981	Meeting in Bethesda, MD, to discuss reformatting of CESSAR Chapter 15, "Accident and Transient Analyses."
June 12, 1981	Letter from CE submitting Amendment No. 3 to the FSAR.
June 23-25, 1981	Meeting in Windsor, CT, to audit the mechanical design of the CESSAR systems and components.
June 25, 1981	Letter from CE identifying schedule for responses to staff request for additional information.
June 26, 1981	Letter to CE requesting Chapter 15 be formatted consistent with the Standard Review Plan and requesting additional information.
July 1, 1981	Meeting in Windsor, CT, to discuss CESSAR/St. Lucie 2 accident analyses.
July 16, 1981	Letter to CE requesting additional information for the CESSAR review.
July 21, 1981	Meeting in Bethesda, MD, to discuss staff questions regarding RCS design.
July 29, 1981	Meeting in Bethesda, MD, to discuss instrumentation and control interfaces.
July 29, 1981	Letter from CE submitting Amendment No. 4 to the FSAR.
July 31, 1981	Letter from CE - documentation of MEB meeting 6/23-25/81.
August 7, 1981	Meeting in Bethesda, MD, to discuss RCS design and accident analyses.
August 14, 1981	Letter from CE submitting transcript from I&C Independent Design Review.
August 20, 1981	Letter from CE providing supporting information for fuel failure criteria.

September 3, 1981	Letter to CE requesting additional information on Unresolved Safety Issues.
September 10, 1981	Meeting in Bethesda, MD, to discuss staff questions on RCS design.
September 11, 1981	Letter to CE requesting additional information for the CESSAR review.
September 17, 1981	Meeting in Bethesda, MD, to discuss preliminary staff positions.
October 2, 1981	Letter from CE submitting proposed amendment for part-loop operation.
October 2, 1981	Letter from CE submitting supporting information on guide tube wear.
October 2, 1981	Meeting in Bethesda, MD, to discuss Reactor Power Cutback System.
October 8, 1981	Letter from CE submitting responses to staff questions and proposed FSAR revisions.
October 14, 1981	Letter from CE submitting supplemental mechanical design information.
October 20, 1981	Meeting in Bethesda, MD, to discuss Environmental Qualification review.
October 26, 1981	Letter to CE requesting additional information on Unresolved Safety Issues.
October 27, 1981	Letter to CE requesting CESSAR quality list be revised.
October 28, 1981	Letter to CE specifying schedule for RPCS review.
October 29, 1981	Letter from CE providing commitment to resolve six CESSAR open items.
October 30, 1981	Letter from CE providing commitment to resolve four CESSAR open items.
November 3, 1981	Letter from CE revising CESSAR quality list.
November 4, 1981	Letter from CE providing commitment to resolve two CESSAR open items.

APPENDIX B

BIBLIOGRAPHY

CODE OF FEDERAL REGULATIONS[1]

10 CFR Part 20, "Standards for Protection Against Radiation"
10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"
10 CFR Part 70, "Domestic Licensing of Special Nuclear Materials"
10 CFR Part 100, "Reactor Site Criteria"
10 CFR Part 140, "Financial Protection Requirements"

GENERAL DESIGN CRITERIA (10 CFR PART 50, APPENDIX A)

1. Quality Standards and Records
2. Design Gases for Protection Against Natural Phenomena
3. Fire Protection
4. Environmental and Missile Design Bases
5. Sharing of Structures, Systems, and Components
10. Reactor Design
11. Reactor Inherent Protection
12. Suppression of Reactor Power Oscillations
13. Instrumentation and Control
14. Reactor Coolant Pressure Boundary
15. Reactor Coolant System Design
16. Containment Design
17. Electric Power Systems
18. Inspection and Testing of Electric Power Systems
19. Control Room
20. Protection System Functions
21. Protection System Reliability and Testability
22. Protection System Independence
23. Protection system Failure Modes
24. Separation of Protection and Control Systems
25. Protection System Requirements for Reactivity Control Malfunctions
26. Reactivity Control System Redundancy and Capability
27. Combined Reactivity Control Systems Capability
28. Reactivity Limits
29. Protection Against Anticipated Operational Occurrences
30. Quality of Reactor Coolant Pressure Boundary
31. Fracture Prevention of Reactor Coolant Pressure Boundary
32. Inspection of Reactor Coolant Pressure Boundary
33. Reactor Coolant Makeup
34. Residual Heat Removal
35. Emergency Core Cooling

[1]Available in public libraries.

36. Inspection of Emergency Core Cooling System
37. Testing of Emergency Core Cooling System
39. Inspection of Containment Heat Removal System
40. Testing of Containment Heat Removal System
41. Containment Atmosphere Cleanup
42. Inspection of Containment Atmosphere Cleanup Systems
43. Testing of Containment Atmosphere Cleanup Systems
44. Cooling Water
45. Inspection of Cooling Water System
46. Testing of Cooling Water System
50. Containment Design Basis
51. Fracture Prevention of Containment Pressure Boundary
52. Capabaility for Containment Leakage Rate Testing
53. Provisions for Containment Testing and Inspection
54. Systems Penetrating Containment
55. Reactor Coolant Pressure Boundary Penetrating Containment
56. Primary Contaiment Isolation
57. Closed System Isolation Valves.
60. Control Releases of Radioactive Materials to the Environment
61. Fuel Storage and Handling and Radioactivity Control
62. Prevention of Criticality in Fuel Storage and Handling
63. Monitoring Fuel and Waste Storage
64. Monitoring Radioactivity Releases

COMBUSTION ENGINEERING REPORTS[2]

CESSAR-F Combustion Engineering Standard Safety Analysis Report - FSAR
CESSAR-P Combustion Engineering Standard Safety Analysis Report - PSAR
CEN-39(a)-P Rev. 2, "CPC Protective Algorithm Software Change Procedure"
CEN-50, "CE Burnable Poison Irraidation Test Program," March 1977
CEN-77(M)-P, "Cladding Damage Analysis of Maine Yankee Core II
CENPD-107 "CESEC--Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," April 1974
CENPD-118 "Densification of Combustion Engineering Fuel," June 1974
CENPD-139-A "Fuel Evaluation Model, July 1974
CENPD-158 "Anticipated Transients Without Scram,"
CENPD-158, Rev. 1
CENPD-162-P-A "Critical Heat Flux Correlation for CE Fuel Assemblies With Standard Spacer Grids, Part 1 Uniform Axial Power Distribution."
CENPD-168 "Design Basis Pipe Breaks"
CENPD-169-P "COLSS--Assessment of the Accuracy of PWR Operating Limits as Determined by the Core Operating Limit Supervisory Systems"
CENPD-178 "Structural Analysis of Fuel Assemblies for Combined Seismic and Loss-of-Coolant Accident Loading," August 1976
CENPD-187-A "CEPAN Method of Analyzing Creep Collapse of Oral Cladding," March 1976
CENPD-190-A "CEA Ejection," January 1976
CENPD-198 (Suppl. 1) "Zircaloy Growth Application of Zircaloy Irradiation Growth Correlations for the Calculation of Fuel Assembly and Fuel Rod Growth Allowances," December 1977

[2]Available for inspection and copying for a fee in the NRC Public Document Room, 1717 H Street, NW., Washington, DC.

CENPD-198 (Suppl. 2-P) "Response to Request for Additional Information on CENPD-198-P, Supplement 1, "November 1, 1978

CENPD-201-A, Supplement 1, "System 80 Reactor Coolant Pump Loss of Component Cooling Water Test Report"

CENPD-207, "Core Thermo-Hydraulics Code"

CENPD-225, "Fuel and Poison Rod Burning," October 1976

CENPD-255, Revision 2, "Classification of IE Instrumentation," August 1981

"Independent Design Review of the System 80 Instrumentation and Control Systems," June 1981

CE Standard Technical Specifications

USNRC REPORTS[3]

WASH-1270	"Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," September 1973.
	"Safety Evaluation of the Comanche Peak Steam Electric Station, Units 1 and 2, "Docket Nos. 50-445 and 50-446, 1974.
	"Safety Evaluation Report Related to Construction of the South Texas Project, Units 1 and 2," Docket Nos. STN 50-498 and 50-499, 1975.
NUREG-75/112	"Safety Evaluation Report for the Preliminary Design Approval of the Combustion Engineering Standard Safety Analysis Report--CESSAR System 80," December 1975
NUREG-0017	"Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents for Pressurized Water Reactors (PWR-GALE Code)," April 1976
NUREG-0085	"The Analysis of Fuel Densification," July 1976
NUREG-0212,	"Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors," December 1980
NUREG-0224	"Final Report on Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors,' September 1978
NUREG-0303	"Evaluation of the Behavior of Water Logged Fuel Rod Failures in LWRs," March 1978
NUREG-0308,	"Safety Evaluation Report Related to Operation of Arkansas Nuclear One, Unit 2," September 1978
NUREG-0347	"Safety Evaluation Report Related to Construction of the Yellow Creek Nuclear Plant," Docket Nos. STN 50-566 and STN 50-567, December 1977

[3]Available for purchase from GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and/or National Technical Information Service, Springfield, VA 22161.

NUREG-0410 "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants, Report to Congress," December 1977

NUREG-0418 "Fission Gas Release From Fuel at High Burnup," March 1978

NUREG-0460 "Anticipated Transients Without Scram for Light Water Reactors," April 1978

NUREG-0510 "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants--A Report to Congress," January 1979

NUREG-0577 "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," September 1979

NUREG-0578 "TMI-2 Lessons Learned Task Force: Status Report and Short-Term Recommendations," July 1979

NUREG-0582 "Water Hammer in Nuclear Power Plants," April 1979

NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," November 1979

NUREG-0609 "Asymmetric Blowdown Loads on PWR Primary Systems, Resolution of Generic Task Action Plan A-2," January 1981

NUREG-0611 "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Pressurized Water Reactors," November 1979

NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," July 1980

NUREG-0630 "Cladding Swelling and Rupture Models for LOCA Analysis," November 1979.

NUREG-0635 "Generic Assessment of Small-Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants," January 1980

NUREG-0649 "Task Action Plan for Unresolved Safety Issues Related to Nuclear Power Plants," February 1980

NUREG-0660, Vols. 1 and 2 "NRC Action Plan Developed as a Result of the TMI-2 Accident," May 1980; Revision 1, August 1980

NUREG-0705 "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants," March 1981

NUREG-0712 "Safety Evaluation Report Related to the Operation of San Onofre, Nuclear Generating Station, Units 2 and 3," February 1981

NUREG-0737 "Clarification of TMI Action Plan Requirements," November 1980

NUREG-0800 (Formerly NUREG-75/087) "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants--LWR Edition," September 1981

USNRC REGULATORY GUIDES[4]

1.1 "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps"

1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors"

1.6 "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems"

1.7 "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident, Rev. 2"

1.8 "Personnel Selection and Training"

1.9 "Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants"

1.11 "Instrument Lines Penetrating Primary Reactor Containment"

1.12 "Instrumentation for Earthquakes"

1.13 "Spent Fuel Stored Facility Design Basis"

1.14 "Reactor Coolant Pump Flywheel Integrity"

1.20 "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing"

1.21 "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Release of Radioactivity in Liquid and Gaseous Effluents From Light Water-Cooled Nuclear Power Plants"

1.23 "Onsite Meteorological Programs, Rev. 1, Sept 1980"

1.25 "Assumptions Used for Evaluating the Potential Radological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors"

1.26 "Quality Group Classifications and Standards for Water-, Steam-, and Radio-waste-Containing Components of Nuclear Power Plants"

1.27 "Ultimate Heat Sink for Nuclear Power Plants"

1.28 "Quality Assurance Program Requirements (Design and Construction)"

[4]Available for purchase from Superintendent of Documents, U.S. Nuclear Regulatory Commission, ATTN: Sales Manager, Washington, DC 20555.

1.29 "Seismic Design Classification"

1.30 "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment"

1.31 "Control of Ferrite Content in Stainless Steel Weld Metal"

1.32 "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants"

1.36 "Nonmetallic Thermal Insulation for Austenitic Stainless Steel"

1.37 "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"

1.38 "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants, Rev. 2"

1.40 "Qualfication Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants"

1.41 "Preoperational Testing of Redundant Onsite Electric Power Systems To Verify Proper Load Group Assignments"

1.43 "Control Stainless Steel Weld Cladding of Low-Alloy Steel Components"

1.44 "Control of the Use of Sensitized Stainless Steel"

1.45 "Reactor Coolant Pressure Boundary Leakage Detection Systems"

1.46 "Protection Against Pipe Whip Inside Containment"

1.47 "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems"

1.48 "Design Limits and Loading Combinations for Seismic Category I Fluid System Components"

1.50 "Control of Preheat Temperature for Welding of Low-Alloy Steel"

1.52 "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants"

1.58 "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel, Rev. 1"

1.59 "Design Basis Floods for Nuclear Power Plants"

1.60 "Design Response Spectra for Seismic Design of Nuclear Power Plants"

1.61 "Damping Values for Seismic Design of Nuclear Power Plants"

1.62 "Manual Initiation of Protective Actions"

1.63 "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants"

1.64 "Quality Assurance Requirements for the Design of Nuclear Power Plants, Rev. 2"

1.67 "Installation of Overpressure Protective Devices"

1.68 "Initial Test Programs for Water-Cooled Nuclear Power Plants"

1.68.2 "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants"

1.69 "Concrete Radiation Shields for Nuclear Power Plants"

1.70 "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 2"

1.71 "Welder Qualification for Areas of Limited Accessibility"

1.74 "Quality Assurance Terms and Definitions"

1.75 "Physical Independence of Electric Systems"

1.76 "Design Basis Tornado for Nuclear Power Plants"

1.77 "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors"

1.79 "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors"

1.80 "Preoperational Testing of Instrument Air Systems"

1.82 "Sumps for Emergency Core Cooling and Containment Spray Systems"

1.83 "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes"

1.84 "Design and Fabrication Code Case Acceptability--ASME Section III, Division 1"

1.85 "Materials Code Case Acceptability--ASME Section III, Division 1"

1.88 "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records, Rev 2"

1.94 "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants, Rev. 1"

1.95 "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release"

1.97 "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident, Rev. 2"

1.102 "Flood Protection for Nuclear Power Plants"

1.106 "Thermal Overload Protection for Electric Motors on Motor-Operated Valves"

1.116 "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems, Rev. 0-R"

1.118 "Periodic Testing of Electric Power and Protection Systems"

1.123 "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants, Rev 1"

1.126 "An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification"

1.133 "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors"

1.141 "Containment Isolation Provisions for Fluid Systems"

1.143 "Design Guidnce for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants"

1.145 "Atmospheric Dispersion Models for Potential Accident Consequence Assessment at Nuclear Power Plants"

TECHNICAL CODES, AND STANDARDS[5]

American Concrete Institute (ACI)

American Institute of Steel Construction (AISC)

American National Standards Institute (ANSI)

American Nuclear Society (ANS)

American Petroleum Institute (API)

American Society of Civil Engineers (ASCE)

American Society of Mechanical Engineers (ASME)

[5]Available from public technical libraries.

American Society of Testing Materials (ASTM)

Institute of Electrical and Electronics Engineers (IEEE)

National Electrical Manufacturers Association (NEMA)

National Fire Protection Association (NFPA)

Welding Research Council

Documents with the following types of designation and other miscellaneous documents are available for inspection and copying for a fee in the NRC Public Document Room at 1717 H Street, NW., Washington, DC:

Commission Order
Inspection and Enforcement documents
Regulatory Guides
Standard Review Plan
Branch Technical Positions

APPENDIX C

NUCLEAR REGULATORY COMMISSION (NRC)
UNRESOLVED SAFETY ISSUES

C.1 Unresolved Safety Issues

The NRC staff continuously evaluates the safety requirements used in its reviews against new information as it becomes available. Information related to the safety of nuclear power plants comes from a variety of sources including experience from operating reactors; research results; NRC staff and Advisory Committee on Reactor Safeguards (ACRS) safety reviews; and vendor, architect/engineer and utility design reviews. Each time a new concern or safety issue is identified from one or more of these sources, the need for immediate action to assure safe operation is assessed. This assessment includes consideration of the generic implications of the issue.

In some cases, immediate action is taken to assure safety, e.g., the derating of boiling water reactors as a result of the channel box wear problems in 1975. In other cases, interim measures, such as modifications to operating procedures, may be sufficient to allow further study of the issue prior to making licensing decisions. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether existing NRC staff requirements should be modified to address the issue for new plants or if backfitting is appropriate for the long term operation of plants already under construction or in operation.

These issues are sometimes called "generic safety issues" because they are related to a particular class or type of nuclear facility rather than a specific plant. These issues have also been referred to as "unresolved safety issues," (NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," dated January 1, 1978). However, as discussed above, such issues are considered on a generic basis only after the staff has made an initial determination that the safety significance of the issue does not prohibit continued operation or require licensing actions while the longer-term generic review is underway.

C.2 ALAB-444 Requirements

These longer-term generic studies were the subject of a Decision by the Atomic Safety and Licensing Appeal Board of the Nuclear Regulatory Commission. The Decision was issued on November 23, 1977 (ALAB-444) in connection with the Appeal Board's consideration of the Gulf States Utility Company application for the River Bend Station, Unit Nos. 1 and 2.

In the view of the Appeal Board,

"In short, the board (and the public as well) should be in a position to ascertain from the SER itself--without the need to resort to extrinsic documents--the staff's perception of the nature and extent of the relationship between each significant unresolved generic safety question and the eventual operation of the reactor under scrutiny. Once again, this assessment might well have a direct bearing upon the ability of the licensing board to make the safety findings required of it on the construction permit level even though the generic answer to the question remains in the offing. Among other things, the furnished information would likely shed light on such alternatively important considerations as whether: (1) the problem has already been resolved for the reactor under study; (2) there is a reasonable basis for concluding that a satisfactory solution will be obtained before the reactor is put in operation; or (3) the problem would have no safety implications until after several years of reactor operation and, should it not be resolved by then, alternative means will be available to insure that continued operation (if permitted at all) would not pose an undue risk to the public."

This appendix is specifically included to respond to the decision of the Atomic Safety and Licensing Appeal Board as enunciated in ALAB-444, and as applied to an operating license proceeding Virginia Electric and Power Company (North Anna Nuclear Power Station, Unit Nos. 1 and 2), ALAB-491, NRC 245 (1978).

C.3 "Unresolved Safety Issues"

In a related matter, as a result of Congressional action on the Nuclear Regulatory Commission budget for Fiscal Year 1978, the Energy Reorganization Act of 1974 was amended (PL 95-209) on December 13, 1977 to include, among other things, a new Section 210 as follows:

"UNRESOLVED SAFETY ISSUES PLAN"

"SEC. 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report of the Commission thereafter."

The Joint Explanatory Statement of the House-Senate Conference Committee for the Fiscal Year 1978 Appropriations Bill (Bill S.1131) provided the following additional information regarding the Committee's deliberations on this portion of the bill:

"SECTION 3 - UNRESOLVED SAFETY ISSUES"

"The House amendment required development of a plan to resolve generic safety issues. The conferees agreed to a requirement that the plan be submitted to the Congress on or before January 1, 1978.

The conferees also expressed the intent that this plan should identify and describe those safety issues, relating to nuclear power reactors, which are unresolved on the date of enactment. It should set forth: "(1) Commission actions taken directly or indirectly to develop and implement corrective measures; (2) further actions planned concerning such measures; and (3) timetables and cost estimates of such actions. The Commission should indicate the priority it has assigned to each issue, and the basis on which priorities have been assigned."

In response to the reporting requirements of the new Section 210, the NRC staff submitted to Congress on January 1, 1978, a report, NUREG-0410, entitled "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," describing the NRC generic issues program. The NRC program was already in place when PL 95-209 was enacted and is of considerably broader scope than the "Unresolved Safety Issues Plan" required by Section 210. In the letter transmitting NUREG-0410 to the Congress on December 30, 1977, the Commission indicated that "the progress reports, which are required by Section 210 to be included in future NRC annual reports, may be more useful to Congress if they focus on the specific Section 210 safety items."

It is the NRC's view that the intent of Section 210 was to assure that plans were developed and implemented on issues with potentially significant public safety implications. In 1978, the NRC undertook a review of over 130 generic issues addressed in the NRC program to determine which issues fit this description and qualify as "Unresolved Safety Issues" for reporting to the Congress. The NRC review included the development of proposals by the NRC Staff and review and final approval by the NRC Commissioners.

This review is described in a report NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants - A Report to Congress," dated January 1979. The report provides the following definition of an "Unresolved Safety Issue:"

> "An Unresolved Safety Issue is a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants it affects."

Further the report indicates that in applying this definition, matters that pose "important questions concerning the adequacy of existing safety requirements" were judged to be those for which resolution is necessary to (1) compensate for a possible major reduction in the degree of protection of the public health and safety, or (2) provide a potentially significant decrease in the risk to the public health and safety. Quite simply, an "Unresolved Safety Issue" is potentially significant from a public safety standpoint and its resolution is likely to result in NRC action on the affected plants.

All of the issues addressed in the NRC program were systematically evaluated against this definition as described in NUREG-0510. As a result, seventeen "Unresolved Safety Issues" addressed by twenty-two tasks in the NRC program were identified. The issues are listed below. Progress on these issues was first discussed in the 1978 NRC Annual Report. The number(s) of the generic task(s) (e.g., A-1) in the NRC program addressing each issue is indicated in parentheses following the title.

"UNRESOLVED SAFETY ISSUES" (APPLICABLE TASK NOS.)

1. Waterhammer - (A-1)
2. Asymmetric Blowdown Loads on the Reactor Coolant System - (A-2)
3. Pressurized Water Reactor Steam Generator Tube Integrity - (A-3, A-4, A-5)
4. BWR Mark I and Mark II Pressure Suppression Containments - (A-6, A-7, A-8, A-39)
5. Anticipated Transients Without Scram - (A-9)
6. BWR Nozzle Cracking - (A-10)
7. Reactor Vessel Materials Toughness - (A-11)
8. Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports - (A-12)
9. Systems Interaction in Nuclear Power Plants - (A-17)
10. Environmental Qualification of Safety-Related Electrical Equipment - (A-24)
11. Reactor Vessel Pressure Transient Protection - (A-26)
12. Residual Heat Removal Requirements - (A-31)
13. Control of Heavy Loads Near Spent Fuel - (A-36)
14. Seismic Design Criteria - (A-40)
15. Pipe Cracks at Boiling Water Reactors - (A-42)
16. Containment Emergency Sump Reliability - (A-43)
17. Station Blackout - (A-44)

In the view of the staff, the "Unresolved Safety Issues" listed above are the substantive safety issues referred to by the Appeal Board in ALAB-444 when it spoke of "... those generic problems under continuing study which have.... potentially significant public safety implications." Thirteen of the 22 tasks identified with the "Unresolved Safety Issues" are not applicable to CESSAR. Six of these thirteen tasks (A-6, A-7, A-8, A-39, A-10 and A-42) are peculiar to boiling water reactors. Tasks A-3 and A-5 address steam generator tube problems in Westinghouse and Babcock and Wilcox plants. The five remaining tasks are outside the scope of CESSAR. The five items listed below are discussed in the plant-specific Safety Evaluation Reports.

TASKS OUTSIDE CESSAR SCOPE

A-1 Water Hammer (BOP design aspects)
A-17 Systems Interaction in Nuclear Power Plants
A-40 Seismic Design Criteria
A-43 Containment Emergency Sump Reliability
A-44 Station Blackout

With regard to the ten applicable items that are inside the scope of CESSAR, the NRC staff has issued NUREG reports providing its proposed resolution to six of these issues. Each of these issues has been addressed in this Safety Evaluation Report or will be addressed in a future supplement. The table below lists these issues and the section of this Safety Evaluation Report in which they are discussed.

Task Number	NUREG Report and Title	Safety Evaluation Report Section
A-2	NUREG-0606, "Asymmetric Blowdown Loads on PWR Primary Systems"	3.9.2
A-9	NUREG-0460, Vol. 4, "Anticipated Transients Without Scram for Light Water Reactors"	15.3.9
A-24	NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"	3.11
A-26	NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors" and RSB BTP 5-2	5.2.2
A-31	SRP 5.4.7 and BTP 5-1, "Residual Heat Removal Systems" incorporate requirements of USI A-31	5.4.3
A-36	NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"	9.1.4

The remaining applicable issues applicable to CESSAR are listed in the following table.

GENERIC TASK ADDRESSING UNRESOLVED SAFETY ISSUES THAT ARE APPLICABLE TO CESSAR

1. A-1 Water Hammer (CESSAR design aspects)
2. A-4 Combustion Engineering Steam Generator Tube Integrity
3. A-11 Reactor Vessel Materials Toughness
4. A-12 Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports

Task Action Plans for the generic tasks above are included in NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants." Draft NUREG-0577 which represents staff resolution of USI A-12 was issued for comment in November 1979. The Draft NUREG contained the Task Action Plan for A-12. The information provided in NUREG-0649 meets most of the informational requirements of ALAB-444. Each Task Action Plan provides a description of the problem; the staff's approaches to its resolution; a general discussion of the bases upon which continued plant licensing or operation can proceed pending completion of the task; the technical organizations involved in the task and estimates of the manpower required; a description of the interactions with other NRC offices, the Advisory Committee on Reactor Safeguards and outside organizations; estimates of funding required for contractor supplied technical assistance; prospective dates for completing the task; and a description of potential problems that could alter the planned approach on schedule.

In addition to the Task Action Plans, the staff issues the "Office of Nuclear Reactor Regulation Unresolved Safety Issues Summary, Aqua Book" (NUREG-0606) on a quarterly basis which provides current schedule information for each of the "Unresolved Safety Issues." It also includes information relative to the implementation status of each "Unresolved Safety Issue" for which technical resolution is complete.

We have reviewed the four "Unresolved Safety Issues" listed above as they relate to CESSAR. Discussion of each of these issues including references to related discussions in the Safety Evaluation Report is provided below in Section C.5. Based on our review of these items, we concluded, for the reasons set forth in Section C.5, that there is reasonable assurance that CESSAR plants can be operated prior to the ultimate resolution of these generic issues without endangering the health and safety of the public.

C.4 New "Unresolved Safety Issues"

An in-depth and systematic review of generic safety concerns identified since January 1979 has been performed by the staff to determine if any of these issues should be designated as new "Unresolved Safety Issues." The candidate issues originated from concerns identified in NUREG-0660, "NRC Action Plan as a Result of the TMI-2 Accident;" ACRS recommendations; abnormal occurrence reports and other operating experience. The staff's proposed list was reviewed and commented on by the ACRS, the Office of Analysis and Evaluation of Operational Data (AEOD) and the Office of Policy Evaluation. The ACRS and AEOD also proposed that several additional "Unresolved Safety Issues" be considered by the Commission. The Commission considered the above information and approved the following four new "Unresolved Safety Issues:"

A-45 Shutdown Decay Heat Removal Requirements

A-46 Seismic Qualification of Equipment in Operating Plants

A-47 Safety Implications of Control Systems

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

A description of the review process for candidate issues, together with a list of the issues considered is presented in NUREG-0705 "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants, Special Report to Congress," dated March 1981. An expanded discussion of each of the new "Unresolved Safety Issues" is also contained in NUREG-0705.

Each of the above items fall outside the scope of CESSAR. These items will be discussed in the reference plant Safety Evaluation Reports.

C.5 Discussion of Tasks as they Relate to CESSAR

The discussion of the following issues will be revised pending receipt of information that is specific to the CESSAR.

A-1 Waterhammer

Waterhammer events are intense pressure pulses in fluid systems caused by any one of a number of mechanisms and system conditions.

Since 1971 there have been over 100 incidents involving waterhammer in pressurized water reactors and boiling water reactors. The waterhammers have involved steam generator feedrings and piping, decay heat removal systems, emergency core cooling systems, containment spray lines, service water lines, feedwater lines and steam lines. However, the systems most frequently affected by waterhammer effects are the feedwater systems. The most serious waterhammer events have occurred in the steam generator feedrings of pressurized water reactors. These types of waterhammer events are addressed in Section 10.4 of this report.

Under Generic Task A-1, the potential for waterhammer in various systems is being evaluated and appropriate requirements and systematic review procedures are being developed to ensure that waterhammer is given appropriate consideration in all areas of licensing review. A technical report, NUREG-0582, "Waterhammer in Nuclear Power Plants" (July 1979), provided the results of an NRC staff review of waterhammer events in nuclear power plants and states current staff licensing positions, completing a major subtask of Generic Task A-1.

With regard to protection against other potential waterhammer events currently provided in plants, piping design codes require consideration of impact loads. Approaches used at the design stage include: (1) increasing valve closure times, (2) piping layout to proclude water slugs in steam lines and vapor formation in water lines, (3) use of snubbers and pipe hangers, and (4) use of vents and drains. In addition, NRC requires that the applicant conduct a preoperational biration dynamic effects test program in accordance with Section III of the ASME Code for all ASME Class 1 and Class 2 piping systems and restraints during startup and initial operation. These tests will provide adequate assurance that the piping and piping restraints have been designed to withstand dynamic effects resulting from valve closures, pump trips, and other operating modes associated with the design operational transients. Water hammer events associated with the secondary system design are outside the scope of CESSAR. They are addressed on a plant specific basis in the individual plant Safety Evaluation Reports.

Nonetheless, in the unlikely event that a large pipe break did result from a severe waterhammer event, core cooling is assured by the emergency core cooling systems and protection against the dynamic effects of such pipe breaks inside and outside of containment is provided.

In the event that Task A-1 identifies some potentially significant waterhammer scenarios which have not explicitly been accounted for in the design and operation of the plant, corrective measures will be required at that time. The task has not as yet identified the need for requiring any additional measures beyond those already implemented.

Based on the foregoing, we have concluded regarding waterhammer events falling within the scope of CESSAR, that CESSAR plants can be operated prior to ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-4 Combustion Engineering Steam Generator Tube Integrity

The primary concern is the capability of steam generator tubes to maintain their inregrity during normal operation and postulated accident conditions.

In addition, the requirements for increased steam generator tube inspections and repairs have resulted in significant increases in occupational exposures to workers. Corrosion resulting in steam generator tube wall thinning (wastage) has been observed in several Westinghouse and Combustion Engineering plants for a number of years. Major changes in the secondary water treatment process essentially eliminated this form of degradation. Another major corrosion-related phenomenon has also been observed in a number of plants in recent years, resulting from a buildup of support plate corrosion products in the annulus between the tubes and the support plates. This buildup eventually causes a diametral reduction of the tubes, called "denting," and deformation of the tube support plates. This phenomenon has led to other problems, including stress corrosion cracking, leaks at the tube/support plate intersections, and U-bend section cracking of tubes which were highly stressed because of support plate deformation.

Specific measures, such as steam generator design features, a secondary water chemistry control and monitoring program, condensate demineralization and condensor tubing material selection, that the applicant has employed to minimize the onset of steam generator tube problems are described in Sections 5.4.2.1 and 10.3.1 of this report. The technical specifications will include requirements for actions to be taken in the event that steam generator tube leakage occurs during plant operation.

Task A-4 is expected to result in improvements in current requirements for inservice inspection of steam generator tubes. These improvements will include a better statistical basis for inservice inspection program requirements and consideration of the cost/benefit of increased inspection. Pending completion of Task A-4, the measures taken at the plant should minimize the steam generator tube problems encountered. Further, the inservice inspection and technical specification requirements will assure that the applicants and the NRC staff are alerted to tube degradation should it occur. Appropriate actions such as tube plugging, increased and more frequent inspections, and power derating could be taken if necessary. Since the improvements that will result from Task A-4 will be procedural, that is, in improved inservice inspection programs, they can be implemented by the applicant at the plant after operation begins, if necessary.

Based on the foregoing, we have concluded that CESSAR plants can be operated prior to ultimate resolution of this generic issues without undue risk to the health and safety of the public

A-11 Reactor Vessel Materials Toughness

Resistance to brittle fracture, a rapidly propagating catastrophic failure mode for a component containing flaws, is described quantitatively by a material property generally denoted as "fracture toughness." Fracture toughness has different values and characteristics depending upon the material being considered. For steels used in a nuclear reactor pressure vessel, three considerations are important. First, fracture toughness increases with increasing temperature; second, fracture toughness decreases with increasing load rates; and third, fracture toughness decreases with neutron irradiation.

In recognition of these considerations, power reactors are operated within restrictions imposed by the technical specifications on the pressure during heatup and cooldown operations. These restrictions assure that the reactor vessel will not be subjected to a combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel materials. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these technical specification limitations.

For the service times and operating conditions typical of current operating plants, reactor vessel fracture toughness for most plants provide adequate margins of safety against vessel failure under operating, testing, maintenance, and anticipated transient conditions, and accident conditions over the life of the plant. However, results from a reactor vessel surveillance program and analyses performed for up to 20 older operating pressurized water reactors and those for some more recent vintage plants show that such vessels will have comparatively short periods of operation. In addition, results from analyses of some reactor vessels may not be maintained in the event that a main steam line break of a loss-of-coolant accident occurs after approximately 20 years of operation. The principal objective of Task A-11 is to develop an improved engineering method and safety criteria to allow a more precise assessment of the safety margins that are available during normal operation and transients in older reactor vessels with marginal fracture toughness and of the safety margins available during accident conditions for all plants.

Since Task A-11 is projected to be completed well in advance of this facility's reactor vessel reaching a fluence level which would noticably reduce fracture resistance, acceptance vessel integrity for the postulated accident conditions will be assured at least until the reactor vessel is reevaluated for long-term acceptability.

In addition, the surveillance program required by 10 CFR Part 50, Appendix H will afford an opportunity to reevaluate the fracture toughness periodically during the first half of design life.

Therefore, based on the foregoing, we have concluded that CESSAR plants can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

A-12 Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports

During the course of the licensing action for North Anna Power Station Unit Nos. 1 and 2, a number of questions were raised as to the potential for lamellar tearing and low fracture toughness of the steam generator and reactor coolant pump support materials for those facilities. Two different steel specifications (ASTM A-36-70a and ASTM A572-70a) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made on those heats for which excess material was available. The toughness of the A36 steel was adequate, but the toughness of the A572 steel was relatively poor at an operating temperature of 80°F.

Since similar materials and designs have been used on other nuclear plants the concerns regarding the supports for the North Anna facilities are applicable to other PWR plants. It was, therefore, necessary to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all operating PWR plants and those in CP and OL review.

NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Support," was issued for comment in November 1979. This report summarizes work performed by the NRC staff and its contractor, Sandia Laboratories, in the resolution of this generic activity. The report describes the technical issues, the technical studies performed by Sandia Laboratories, the NRC staff's technical positions based on these studies, and the NRC staff's plan for implementing its technical positions. As a part of initiating the implementation of the findings in this report, letters were sent to all applicants and licensees on May 19 and 20, 1980. In these letters a revised proposed implementation plan was presented and specific criteria for material qualifications were defined.

Many comments on both the draft of NUREG-0577 and the letters of May 19 and 20 have been received by the NRC staff and detailed consideration is presently being given to these comments. After completing our review and analysis of the comments provided, we will issue the final revision of NUREG-0577 which will include a full discussion and resolution of the comments and a final plan for implementation.

We estimate that our implementation review will require approximately two years. Since many factors (initiating event, low fracture toughness in a critical support member in tension, low operating temperature, large flaw) must be simultaneously present for failure of the support system we have determined that licensing for pressurized water reactors should continue during the implementation phase. Our conclusions regarding licensing and subsequent operation are not sensitive to the estimated length of time required for this work.

With regard to the lamellar tearing issue, the results of an extensive literature survey by Sandia revealed that, although lamellar tearing is a common occurrence in structural steel construction, virtually no documentation exists describing inservice failures due to lamellar tearing. Nonetheless, additional research is recommended to provide a more definitive and complete evaluation of the importance of lamellar tearing to the structural integrity of nuclear power plant support systems.

Based on our review, we have concluded that there is reasonable assurance that CESSAR plants can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

APPENDIX D

ABBREVIATIONS

ACCWS	- auxiliary component cooling water system
ACFM	- actual cubic feet per minute
ACRS	- Advisory Committee on Reactor Safeguards
ADV	- atmospheric dump valve
AFW	- auxiliary feedwater
AFWS	- auxiliary feedwater system
ALARA	- as low as reasonably achievable
ANO-2	- Arkansas Nuclear One, Unit 2
ANSI	- American National Standards Institute
ASB	- Auxiliary Systems Branch
ASI	- axial shape index
ASLB	- Atomic Safety and Licensing Board
ASME	- American Society of Mechanical Engineers
ANS	- American Nuclear Society
ATWS	- anticipated transient without scram
B&O	- Bulletins and Orders (Task Force)
BOL	- beginning of life
BTP	- Branch Technical Position
CACS	- containment air cooler system
CAD	- containment atmosphere dilution
CARS	- containment atmosphere release system
CCAS	- containment cooling actuation signal
CCS	- containment cooling system
CCW	- component cooling water
CE	- Combustion Engineering, Inc.
CEA	- control element assembly
CEADS	- control element assembly drive system
CEDM	- control element drive mechanism
CFR	- Code of Federal Regulations
CESSAR	- Combustion Engineering Standard Safety Analysis Report
CHF	- critical heat flux
CGCS	- combustible gas control system
CIAS	- containment isolation actuation signal
CIS	- containment isolation signal
COLSS	- core operating limit supervisory system
CP	- construction permit
CPC	- core protection calculator system
CPIS	- containment purge isolation signal
CRD	- control rod drive
CRT	- cathode-ray tube
CSAS	- containment spray actuation system
CSS	- containment spray system
CVAS	- controlled ventilation area system

CVCS - chemical and volume control system
CVN - Charpy V-notch
CWS - circulating water system
DBA - design basis accident
DE - dose equivalent
DEG - double-ended guillotine
DES - double-ended slot
DNB - departure from nucleate boiling
DNBR - departure from nucleate boiling ratio
DOE - Department of Energy
EAB - exclusion area boundary
EAL - emergency action level
ECCS - emergency core cooling system
EFAS - emergency feedwater actuation system
EFW - emergency feedwater
EFS - emergency feedwater system
EHC - electrohydraulic control
EMI - electrical magnetic interference
EOL - end of life
EPZ - emergency planning zone
ESF - engineered safety feature
ESFAS - engineered safety feature actuation signal
ESWS - essential service water system
FDA - Final Design Approval
FHBVS - fuel handling building ventilation system
FHS - fuel handling system
FMEA - failure modes and effects analysis
FSAR - Final Safety Analysis Report
GDC - General Design Criterion
HAZ - heat-affected zone
HELB - high energy line break
HEPA - high-efficiency particulate air
HID - high impact design
HPCI - high-pressure coolant injection
HPSI - high-pressure safety injection
H&V - heating and ventilating
HVAC - heating, ventilation, and air conditioning
I&C - instrumentation and control
ICC - inadequate core cooling
ICSB - Instrumentation and Control Systems Branch
IDR - Independent Design Review
IE - Office of Inspection and Enforcement
IEEE - Institute of Electrical and Electronics Engineers
INEL - Idaho National Engineering Laboratories
ISEG - independent safety engineering group
ISI - inservice inspection
LCO - limiting conditions of operation
LER - licensee event report
LLL - Lawrence Livermore Laboratory
LOCA - loss-of-coolant accident
LOVS - loss-of-voltage signal
LPCI - low-pressure coolant injection
LPD - local power density

LPZ	-	low population zone
LPSI	-	low-pressure safety injection
LPMS	-	loose parts monitoring system
LTOP	-	low temperature overpressure protection
LWR	-	light water reactor
MCC	-	motor control center
MEV	-	million electron volt
MFIV	-	main feedwater isolation valve
MSIS	-	main steam isolation signal
MSIV	-	main steam isolation valve
MSLB	-	main steam line break
MSSS	-	main steam supply system
MTEB	-	Materials Engineering Branch
MVA	-	million volt amp
MWe	-	megawatts (electrical)
MWt	-	megawatts (thermal)
NDE	-	nondestructive examination
NEMA	-	National Electrical Manufacturers Association
NFPA	-	National Fire Protection Association
NIS	-	nuclear instrumentation system
NNS	-	non-nuclear safety
NPSH	-	net positive suction head
NRC	-	U.S. Nuclear Regulatory Commission
NRR	-	Office of Nuclear Reactor Regulation
NSSS	-	nuclear steam supply system
OBE	-	operating basis earthquake
ORNL	-	Oak Ridge National Laboratory
PCB	-	power circuit breaker
PCI	-	pellet/cladding interaction
PCS	-	primary coolant system
PD	-	pump discharge
PMF	-	probable maximum flood
PORV	-	power operated relief valve
PPS	-	plant protection system
PSAR	-	preliminary safety analysis report
PWR	-	pressurized water reactor
QA	-	quality assurance
QAC	-	Quality Assurance Criteria
RAB	-	reactor auxiliary building
RAS	-	recirculation actuation signal
RCP	-	reactor coolant pump
RCPB	-	reactor coolant pressure boundary
RCS	-	reactor coolant system
RG	-	Regulatory Guide
RHR	-	residual heat removal
RMS	-	radiation monitoring system
RPS	-	reactor protection system
RPV	-	reactor pressure vessel
RTS	-	reactor trip system
RTSS	-	reactor trip switchgear system
RWCUS	-	reactor water cleanup system
RWSP	-	refueling water storage pool
RWST	-	refueling water storage tank

SAR - safety analysis report
SBVS - shield building ventilation system
SCA - single-channel analyzer
SCC - stress corrosion cracking
SDC - shutdown cooling
SDCS - shutdown cooling system
SER - safety evaluation report
SGTR - steam generator tube rupture
SIAS - safety injection actuation signal
SIS - safety injection system
SLB - steam line break
SONGS-2&3 - San Onofre Nuclear Generating Station, Units 2 and 3
SPDS - safety parameter display system
SPS - Supplementary Protection System
SQRT - Seismic Qualification Review Team
SRP - Standard Review Plan
SSE - safe shutdown earthquake
STP - standard temperature and pressure
STS - Standard Technical Specifications
TMI-2 - Three Mile Island Unit 2
TSP - trisodium phosphate
UHS - ultimate heat sink
UT - ultrasonic inspection

APPENDIX E

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The Safety Evaluation Report for the application filed by Combustion Engineering, Inc. for a Final Design Approval for the Combustion Engineering Standard Safety Analysis Report (STN 50-470) has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. This report summarizes the results of the staff's safety review of the Combustion Engineering standard nuclear steam supply system. Subject to favorable resolution of items discussed in the Safety Evaluation Report, the staff concludes that the facilities referencing CESSAR, subject to approval of the balance-of-plant design, can conform with the provisions of the Act and the rules and regulations of the Commission.

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