NUREG-0189

Safety Evaluation Report

U. S. Nuclear
Regulatory Commission

related to construction of

Office of Nuclear
Reactor Regulation

Cherokee Nuclear Station, Units 1, 2 and 3

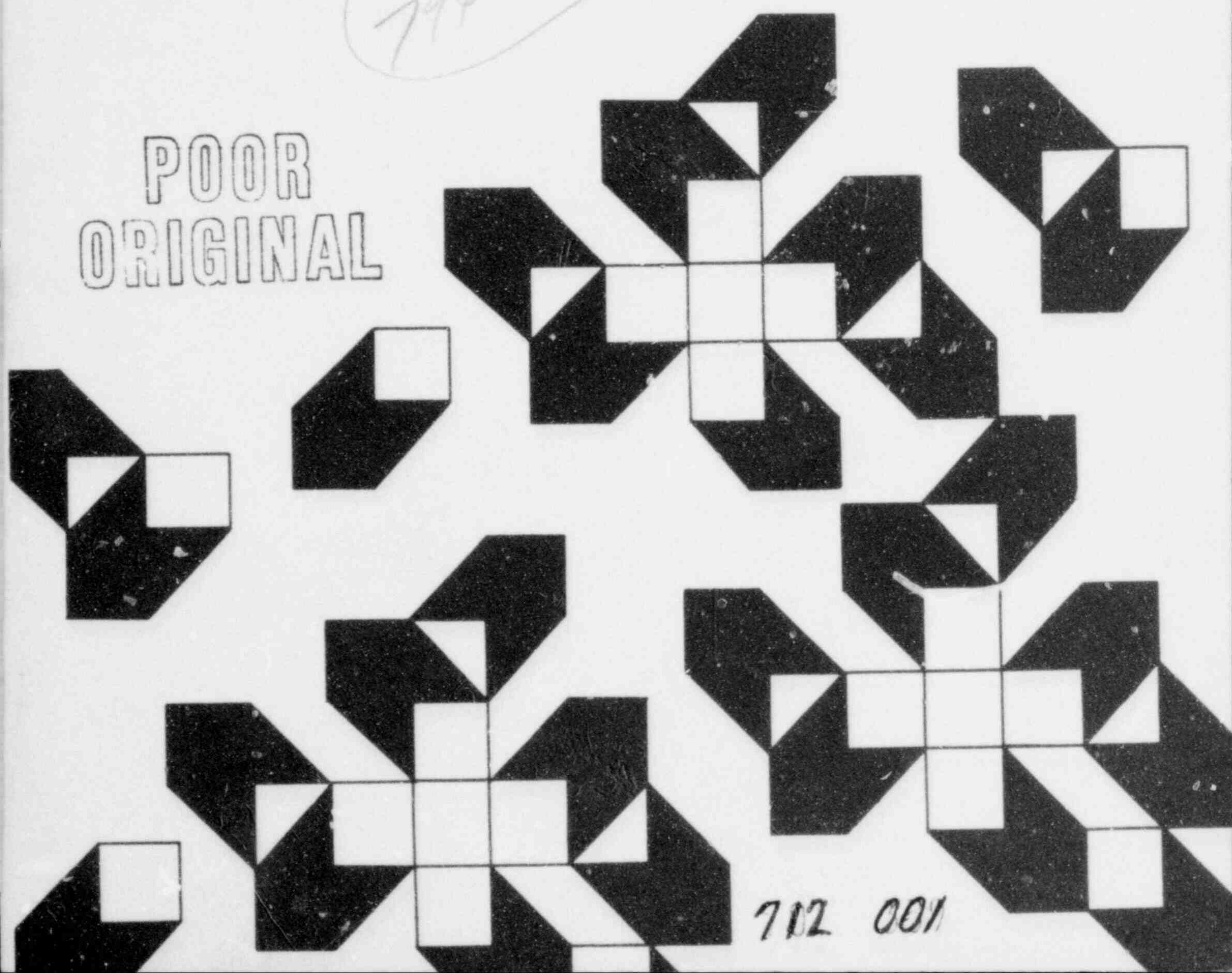
Docket Nos. STN 50-491
STN 50-492
STN 50-493

Duke Power Company

March 1977

7909040087

POOR ORIGINAL



712 001

Available from
National Technical Information Service
Springfield, Virginia 22161
Price: Printed Copy $10.75 ; Microfiche $3.00

NUREG-0189
March, 1977

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

DUKE POWER COMPANY

CHEROKEE NUCLEAR STATION

UNITS 1, 2 AND 3

DOCKET NOS. STN 50-491, STN 50-492 AND STN 50-493

TABLE OF CONTENTS

712 004

TABLE OF CONTENTS (Continued)

712 005

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

712 009

TABLE OF CONTENTS (Continued)

712 010

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

712 012

APPENDICES

712 013

LIST OF TABLES

712 014

LIST OF FIGURES

1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Duke Power Company (hereinafter refered to as the applicant) filed with the Nuclear Regulatory Commission (NRC or Commission) an application docketed on May 24, 1974, for licenses to construct and operate its proposed Cherokee Nuclear Station Units 1, 2, and 3 (Cherokee Nuclear Station or facility). The facility will be located on a 1560-acre site adjacent to the Broad River in the eastern portion of Cherokee County, South Carolina, approximately eight miles southwest of Gaffney, South Carolina, the county seat, 21 miles east of Spartanburg, South Carolina, and 40 miles southwest of Charlotte, North Carolina. The Cherokee Nuclear Station will utilize a Combustion Engineering, Incorporated, standard reference nuclear steam supply system. The application for the Cherokee facility also served as an application for the applicant's proposed Perkins Nuclear Station, Units 1, 2 and 3. A separate Environmental Report was submitted for each facility in accordance with the Commission's regulation, 10 CFR Part 51, which implements the National Environmental Policy Act of 1969 (NEPA).

A single Preliminary Safety Analysis Report (PSAR) was submitted with the application. This PSAR is titled, "Duke Power Company, Project 81, Preliminary Safety Analysis Report." Major portions of this report describe the design of the balance of plant structures, systems and components and incorporates, by reference, sections of the Combustion Engineering report, "Combustion Engineering Standard Safety Analysis Report" (CESSAR). The CESSAR describes the design of the System 80 nuclear steam supply system. The same System 80 nuclear steam supply system design will be used for both facilities. Also, designs of major balance of plant structures, systems and components will be identical. Designs of other structures, systems and components will be functionally identical but may have different configurations to adapt to different site features. For such differences, and for site characteristic descriptions, data, and analyses, two sections of the PSAR are provided (blue paper for Cherokee and pink paper for Perkins).

The Commission issued WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants," on August 20, 1974. Amendment 1 to WASH-1341, dealing with "options" and "overlaps" was issued on January 16, 1975. The regulations governing the submittal and review of standard designs under the "reference system" option are found in Appendix O to Part 50, "Licensing of Production and Utilization Facilities," and Section 2.110 of Part 2, "Rules of Practice" of Title 10 of the Code of Federal Regulations (CFR).

The CESSAR was submitted by Combustion Engineering in the form of an application for a Preliminary Design Approval from the Commission and was in response to Option 1 of

712 016

the Nuclear Regulatory Commission's standardization policy. Option 1 allows for the review of a "reference system" that involves an entire facility design or major fraction of a design outside the context of a license application. On December 19, 1973, the application for the CESSAR was docketed.

Our evaluation of the CESSAR is presented in our Safety Evaluation Report "Safety Evaluation Report by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, In the Matter of Combustion Engineering, Incorporated, CESSAR System 80, Docket No. STN 50-470" NUREG-75/112, December 31, 1975. A copy of NUREG-75/112 is attached as Appendix A to this report, and we have referenced sections of Appendix A in this report as appropriate.

Duke Power Company's application was for licenses to construct and operate its proposed Cherokee and Perkins facilities. Our evaluation of the application for the Perkins facility will be presented in a separate Safety Evaluation Report. Where reference is made herein to the Cherokee PSAR, the reference is to the portions of the Project 81 PSAR common to both the Cherokee and Perkins facilities and to portions of the Project 81 PSAR applicable only to the Cherokee facility. The applicant states in Section 1.1 of the PSAR that the Perkins Nuclear Station and the Cherokee Nuclear Station are intended to be duplicates. Although we conducted a single review of common features and a concurrent review of site related matters for the two facilities, we are issuing separate and complete Safety Evaluation Reports for each of the two stations and the Commission will make two separate decisions on the issuance of construction permits, i.e., one decision for three permits for the proposed Cherokee Nuclear Station and one decision for three permits for the proposed Perkins Nuclear Station.

The information in the PSAR was supplemented by Amendments 1 through 28. Copies of the PSAR, as amended, are available for public inspection at the U.S. Nuclear Regulatory Commission's Public Document Room, 1717 H Street, NW, Washington, D. C. 20555, and at the Cherokee County Library, 300 East Rutledge Avenue, Gaffney, South Carolina 29340.

This Safety Evaluation Report summarizes the results of the technical evaluation of the proposed Cherokee Nuclear Station performed by the Commission's staff and delineates the scope of the technical matters considered in evaluating the radiological safety aspects of the facility. Aspects of the environmental impact considered in the review of the facility in accordance with 10 CFR Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection" of the Commission's regulations, which implements the National Environmental Policy Act of 1969, are discussed in the Commission's Final Environmental Statement which was issued in October 1975.

Upon favorable resolution of the outstanding issues discussed herein and summarized in Section 1.9 of this report, we will be able to conclude that the Cherokee Nuclear Station can be constructed and operated as proposed without endangering the health and safety of the public. Our detailed conclusions are presented in Section 21.0 of this report.

The review and evaluation of the proposed design of the facilities reported herein is only the first stage of a continuing review by the Commission's staff of the design, construction, and operating features of the Cherokee facility. Construction will be accomplished under the surveillance of the Commission's staff. Prior to issuance of operating licenses, we will review the final design to determine that all of the Commission's safety requirements have been met. The facility may then be operated only in accordance with the terms of the operating licenses and the Commission's regulations under the continued surveillance of the Commission's staff.

1.2 General Plant Description

1.2.1 Reference System Design Scope

The CESSAR reference system design scope, as stated in Section 1.2 of Appendix A to this report, comprises the following systems:

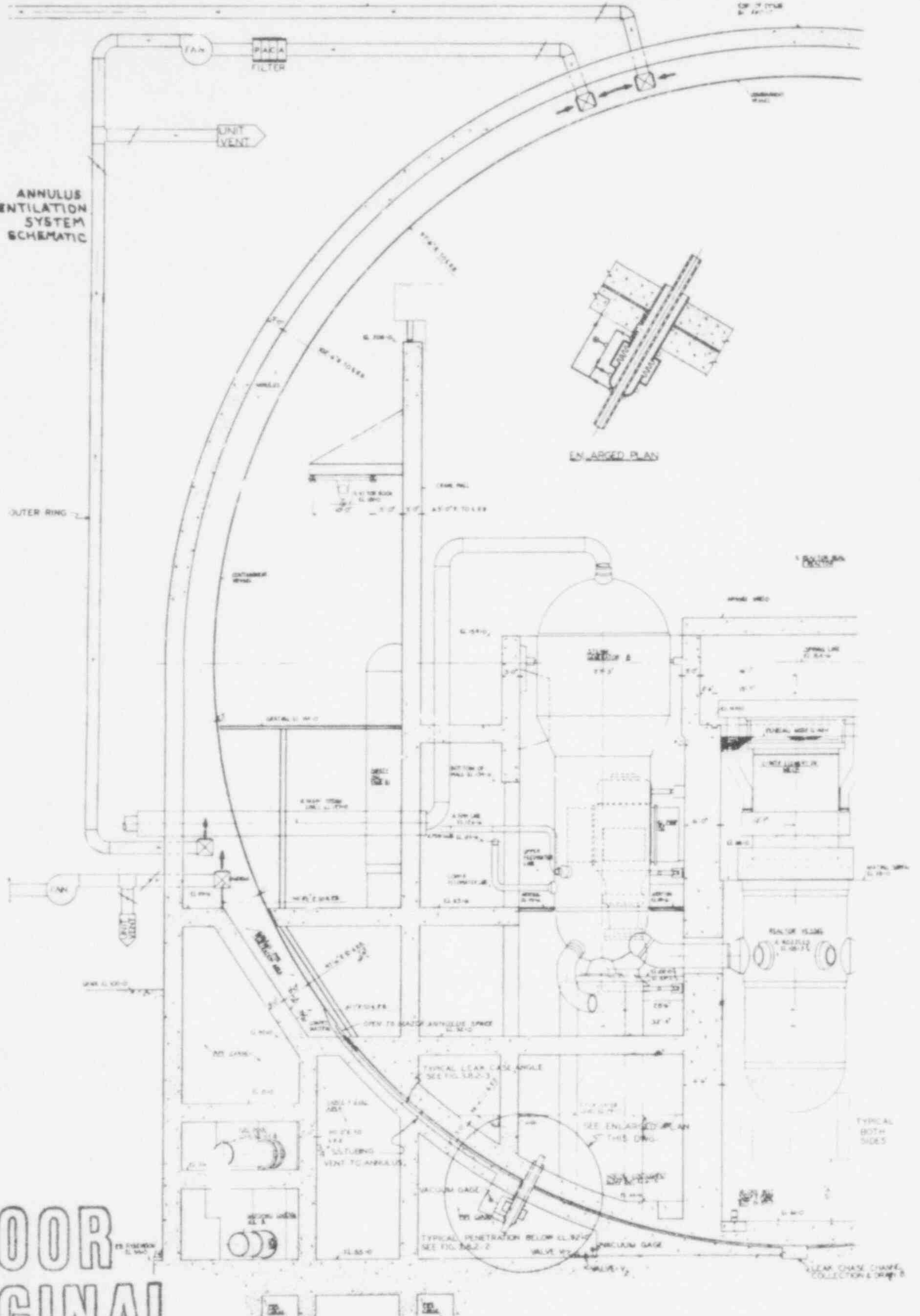
(1) Reactor system
(2) Reactor coolant system
(3) Reactor control system
(4) Reactor protective system
(5) Engineered safety features actuation system
(6) Chemical and volume control system
(7) Shutdown coolant system
(8) Safety injection system
(9) Fuel handling system

A summary description of each of these nine systems is presented in Sections 1.2.1 through 1.2.9 of Appendix A to this report.

1.2.2 Containment and Shield Building

The containment as shown in Figure 1.1 will be a 195-foot diameter spherical steel shell with a wall thickness of one and five-eighths inches. This containment shell will be supported in but not anchored to a spherical depression in an intermediate floor of the shield building, which in the application is also referred to as the reactor building. The shield building will be a reinforced concrete cylindrical building with a spherical dome, that totally encloses the containment. The outer periphery of the containment support floor is at plant elevation 92.0 feet relative to a plant grade elevation of 100.0 feet. All postulated containment leakage following postulated accidents will be collected in the annulus above elevation 92.0 feet, either by direct leakage into the annulus above elevation 92.0 feet or through a leak chase, or collection, system consisting of a network of steel channels welded over containment welds and penetration seal welds as shown in Figure 1.2.

An annulus ventilation system will continuously circulate air from the annulus through engineered safety features filter systems at a rate of about 16,000 cubic feet per minute for each redundant train after a vacuum of about 0.5 inches of water gauge is drawn by exhausting air from the annulus through the plant vent during the first 80 seconds following a postulated loss-of-coolant accident.



POOR ORIGINAL

Figure 1.1 Containment and Shield Building

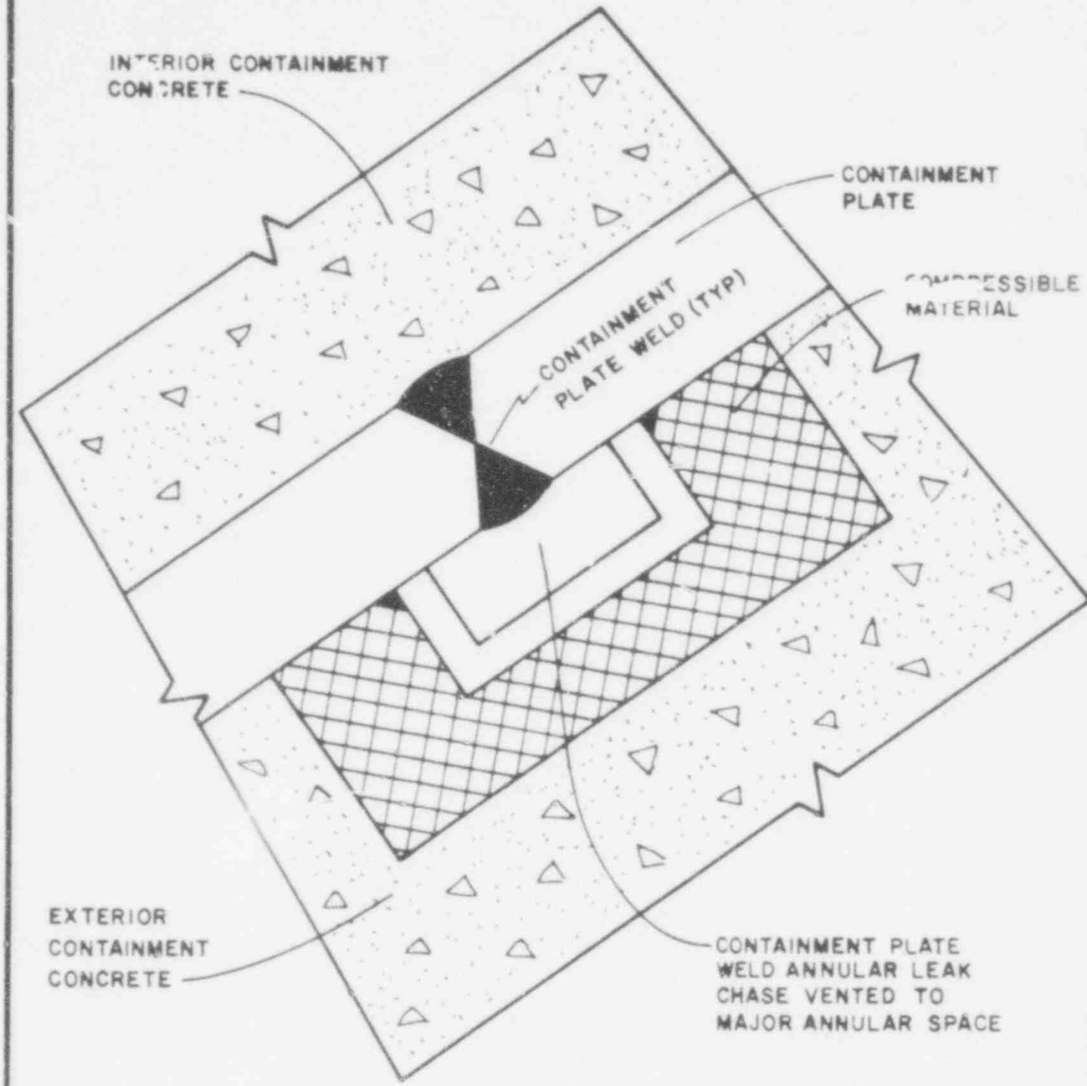


FIGURE 1.2
LOWER CONTAINMENT PLATE WELD ANNULAR LEAK DETAIL CHASE

After the vacuum is achieved, air would be exhausted at a rate of 400 cubic feet per minute or less to match the inflow to the annulus consisting of outward containment leakage, inward leakage through the shield building and upward leakage through the containment support floor.

Space below the containment and inside the shield building will be occupied by engineered safety features equipment, e.g., emergency core cooling system equipment, containment spray system equipment, and shutdown coolant system equipment. Some of the containment penetrations terminate in those areas below the containment and others pass through the annulus above elevation 92.0 feet and terminate outside the shield building. Since the containment support floor is not a fluid seal, postulated, but unlikely, pipe breaks in the regions below the containment could result in external pressures on the containment. The containment will be designed to withstand these pressures without utilization of vacuum relief devices. Guard pipes will be provided around high energy lines that traverse the annulus. Although similarly unlikely to occur, cracks in moderate energy lines within the annulus could be a potential source of water that could flood the containment support floor and, depending on the existing leak characteristics, could flood the spaces below the containment support floor. The facility will be designed such that these effects will be prevented from impairing the function of the containment and other engineered safety features.

1.2.3 Other Major Structures

The shield building as is shown in Figure 1.3 has immediately adjacent to it the auxiliary building, which includes areas for fuel handling, auxiliary equipment, the control room, and a non-seismic Category I control annex that will be supported on portions of the seismic Category I auxiliary building. The end of the turbine building will abut this control annex such that an extension of the turbine generator axis will pass through the center of the containment. Each of the three units is identical, and the turbine generator axes are parallel and about 400 feet apart.

Three circular mechanical draft cooling towers will be located about 900 feet east of the Unit 3 shield building and six like towers will be located about 800 feet west of the Unit 1 shield building. Two small cooling towers will be located just north of these six primary cooling towers to reject heat from the nuclear service water system. Makeup to the nine main towers and the two nuclear service water cooling towers will be provided by pumping water from the makeup intake structure located in an intake sedimentation basin, adjacent to the Broad River.

The intake sedimentation basin will be replenished by water pumped from the river intake structure located in the Broad River. Two nuclear service water pump structures will be located between the nuclear service water cooling towers and Unit 1. Each will house three pumps, one for each unit, to pump water to a component cooling water heat exchanger in one of two component cooling loops for each unit.

Figure 1.3 Facility and Site Features

POOR ORIGINAL

712 022

1.2.4 Ultimate Heat Sink

An alternate nuclear service water pond south of the six cooling towers will serve as a backup source of makeup to the nuclear service water cooling towers during shutdown cooling. A large pond will be located east of the six cooling towers and will be connected by underground pipes to the nuclear service water pump structures. Water pumped from these structures through underground pipes to each unit can be discharged through underground pipes back to the cooling pond. The complex of the two ponds and two cooling towers has be identified as the ultimate heat sink. The ultimate heat sink will be designed to provide facility cooling capability under various conditions of severe natural phenomena and failures of man-made structures.

1.2.5 Permanent Dewatering System

The external walls of the complex of the shield building and the auxiliary building will be surrounded by a vertical wall drain system and located over an underdrain channel system. As shown in Figure 1.4 each system will be connected by an independent set of redundant pipes to a sump located in the auxiliary building. Pumps in that sump will be operated to depress the water level around the buildings to the foundation level and to alleviate the rebound pressure in the foundation rocks below the buildings. Water pumped from the sump will be discharged in the yard adjacent to the auxiliary building to flow by gravity through the yard storm drain system to an auxiliary holding pond north of the plant adjacent to the Broad River.

Under normal conditions, the permanent dewatering system would lower the groundwater level out to a "radius of influence" about 200 feet from the wall drains. There are potentials for increased flow rates into the sump during periods of heavy precipitation and failures of fluid systems within the radius of influence. As indicated in Section 2.4.5, 3.1, and 9.58, the applicant will make measurements of groundwater flows during construction and in the final design will account for the effects of natural phenomena and postulated accident conditions. In addition, the permanent dewatering system and auxiliary holding pond will be elements in the liquid pathway to the Broad River following postulated failures of liquid radwaste tanks. Some of the construction details that will be discussed in Sections 2.4 and 2.5 are shown in Figure 1.5.

1.3 Shared Systems

Systems and components within the scope of the standard reference system that are important to safety will not be shared (Section .2 of Appendix A to this report). Within the Cherokee facility, sharing of structures, systems and components among the three units is limited to the (1) switchyard, telemetering and load dispatch equipment, (2) the ultimate heat sink including ponds and cooling tower structure, (3) intake structure for the nuclear service water system, (4) makeup and blowdown systems for the condenser cooling towers, (5) onsite and offsite environmental monitoring systems,

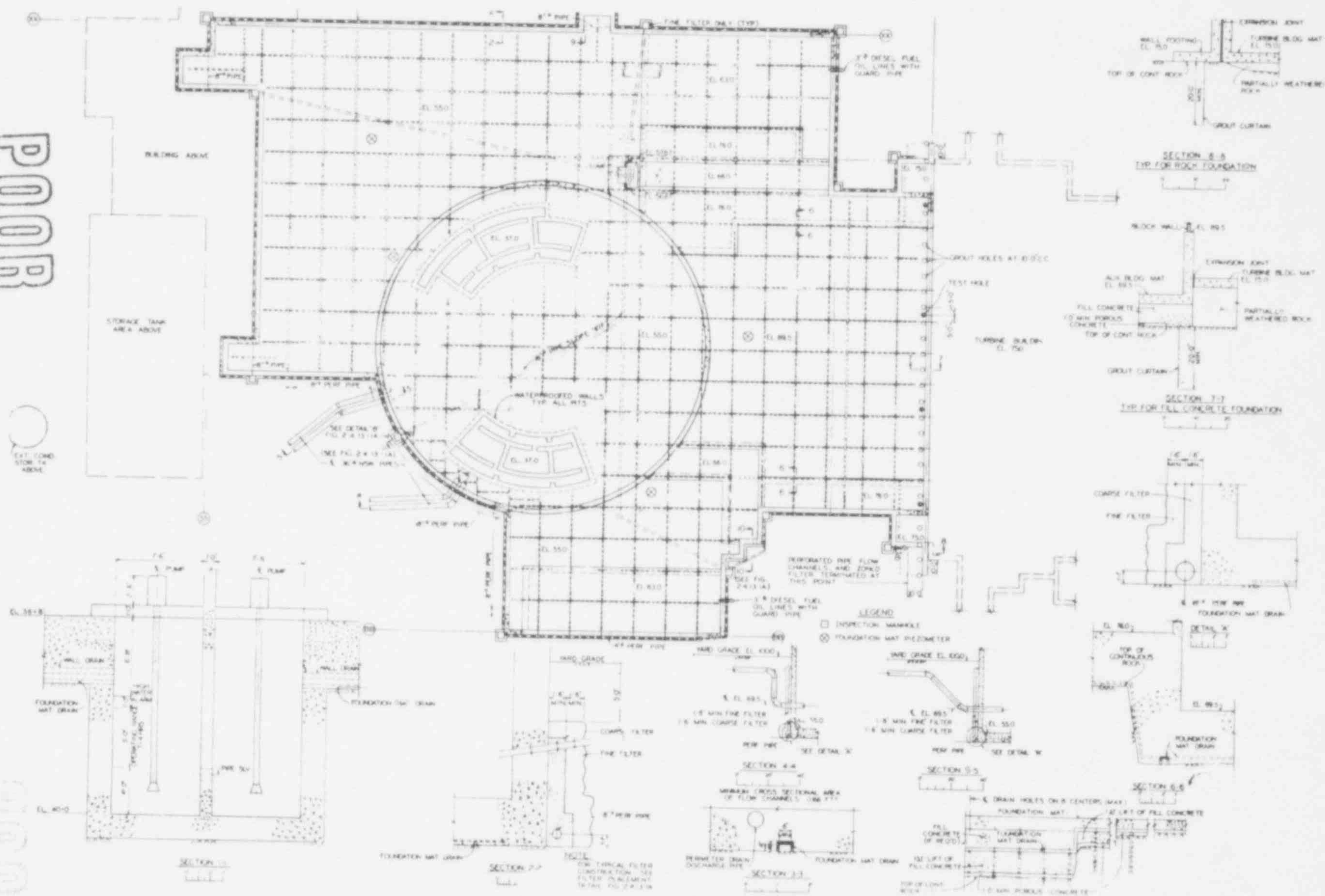


Figure 1.4 Permanent Dewatering System

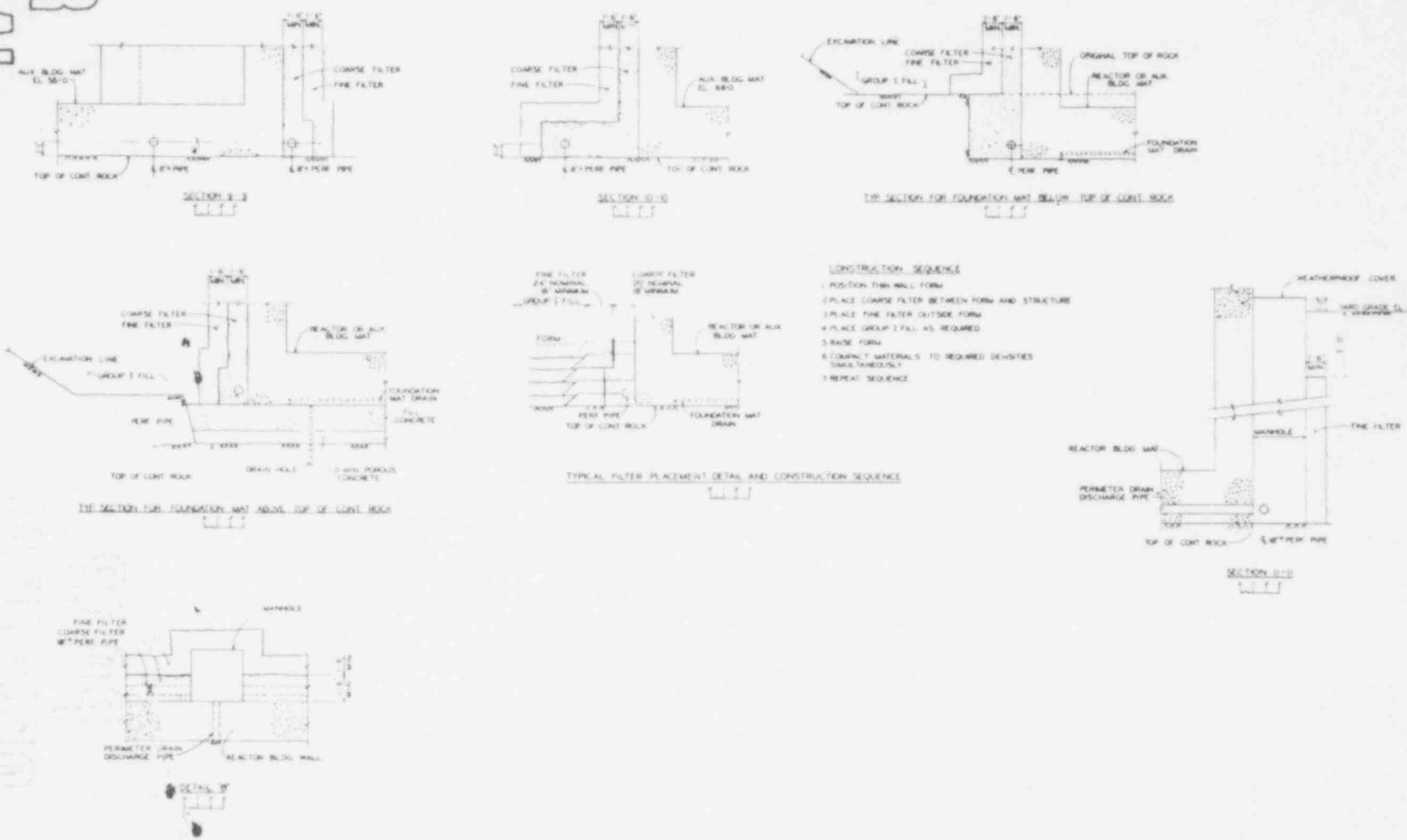


Figure 1.5 Construction Procedures for Permanent Dewatering System

POOR ORIGINAL

7912 025

(6) the fire protection system, (7) the service building, (8) the compressed air system, and (9) underground Class IE electric cable tunnels. Our review has considered sharing of facilities and it is discussed in appropriate sections of this report.

1.4 Comparison with Similar Facility Designs

Some features of the CESSAR are new Combustion Engineering designs. However, many aspects of the plant are similar to those we have evaluated and previously approved for other nuclear power plants. To the extent feasible and appropriate, we have made use of our previous evaluations during our review of those features that are similar to CESSAR features. Where this has been done, the appropriate sections of Appendix A to this report identify the specific applications involved.

To assist in better understanding the relationship of the System 80 reference system design, as described in the CESSAR, to other Combustion Engineering designs, we have presented a comparison of System 80 and San Onofre Units 2 and 3 (Docket Nos. 50-361 and 50-362) principal design features in Table 4.1 of Appendix A to this report. The Safety Evaluation Reports for the applications mentioned in Appendix A to this report are available for public inspection in the Nuclear Regulatory Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., 20555.

1.5 Identification of Agents and Contractors

The Duke Power Company is the applicant for the construction permits for the Cherokee facility, and is responsible for the design, construction and operation of the units.

The applicant engaged Combustion Engineering, Incorporated, to design, manufacture, and deliver to the site the nuclear steam supply system and initial core for each unit. Combustion Engineering will also provide technical assistance during the erection of each nuclear steam supply system, core loading, startup, and preoperational testing. The applicant will perform the architectural engineering and construction services. The turbine-generator will be manufactured by the General Electric Company.

The applicant will also employ consultants, as required, in specialized areas; for example, Law Engineering Testing Company is assisting in the seismologic and geologic studies.

1.6 Summary of Principal Review Matters

Our technical review and evaluation of the information submitted by the applicant considered the principal matters summarized below:

(1) We evaluated the population density and land use characteristics of the site environs and the physical characteristics of the site (including seismology, meteorology, geology, and hydrology) to establish that these characteristics

have been determined adequately and have been given appropriate consideration in the plant design, and that the site characteristics are in accordance with the Commission's siting criteria (10 CFR Part 100), taking into consideration the design of the facilities, including the engineered safety features provided.

(2) We have evaluated the design, fabrication, construction and testing criteria, and expected performance characteristics of the plant structures, systems, and components important to safety to determine that they are in accord with the Commission's General Design Criteria, Quality Assurance Criteria, Regulatory Guides, and other appropriate rules, codes and standards, and that any departure from these criteria, codes and standards have been identified and justified.

(3) We evaluated the expected response of the facilities to various anticipated operating transients and to a broad spectrum of postulated accidents. Based on this evaluation, we determined that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered. We performed conservative analyses of these design basis accidents 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.

(4) We evaluated the applicant's engineering and construction organization, plans for the conduct of plant operations (including the organizational structure and the general qualifications of operating and technical support personnel), the plans for industry security, and the planning for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that the applicant will be technically qualified to safely operate the facilities.

(5) We evaluated the design of the systems provided for control of the radiological effluents from the facilities to determine that these systems will be capable of controlling the release of radioactive wastes from the facility within the limits of the Commission's regulations (10 CFR Part 20), and that the equipment to be provided will be capable of being operated by the applicant in such a manner as to reduce radioactive releases to levels that are as low as reasonably achievable within the context of the Commission's regulations (10 CFR Part 50), and to meet the dose design objectives of Appendix I, 10 CFR Part 50.

(6) We evaluated the applicant's quality assurance program for the design and construction of the facilities to assure that the program complies with the intent of the Commission's regulations (10 CFR Part 50) and that the applicant will have proper controls over the facility design and construction such that there will be a high degree of assurance that, when completed, the facilities can be operated safely and reliably.

(7) We are evaluating the financial data and information supplied by the applicant as required by the Commission's regulations (Section 50.33(f) of 10 CFR Part 50

and Appendix C to 10 CFR Part 50) to determine that the applicant is financially qualified to design and construct the proposed facilities. We will report the results of our evaluation in a supplement to this report.

1.7 Facility Modifications as a Result of Staff Review

During the review of the application for the Cherokee Nuclear Station, numerous meetings were held with the applicant's representatives, its contractors, and consultants to discuss the proposed facility and the technical material submitted. A chronological listing of the meetings and other significant events is given in Appendix B to this report. During the course of the review, the applicant proposed or we requested a number of technical and administrative changes. These are described in various amendments to the original application, and are discussed in appropriate sections of this report.

1.8 Requirements for Future Technical Information

Our evaluation of the requirements for future technical information within the scope of the CESSAR is presented in Section 1.4 of Appendix A to this report. We have also identified the need for certain design information that we normally review before the applicant begins the construction of certain structures in the event of a favorable decision on issuance of construction permits. Since most of this information will be obtained from the site during construction, and since interpretation and judgement may be required to develop detailed procedures for subsequent construction actions, e.g., placement of compacted fill, we conclude that our review of these matters should be made during construction rather than later during the operating license stage of review. The applicant has committed to providing the following information during construction, if construction permits are granted:

(1) Details of the nuclear service water pond overflow spillway for our review and approval prior to construction (Section 2.4.2(4)).

(2) Details of the alternate nuclear service water makeup pond for our review and approval prior to construction (Section 2.4.2(5)).

(3) Details of foundation preparation, including blasting controls, control of engineered fill and inspection of excavation for nuclear service water dam, and of settlement monitoring (Section 2.5.3).

(4) Details of the geologic mapping program during excavation (Appendix F to this report).

(5) Details of the implementation of the program to environmentally qualify Class IE electrical equipment (Section 3.11).

712 028

1.9 Outstanding Items

We have identified certain outstanding issues in our review, some of which will require that the applicant provide additional information to confirm that the proposed design will meet our requirements. Items 1 through 9 are issues that require additional information. Items 10 through 12 are issues where we are currently reviewing information provided by the applicant, and where our review is not yet complete. These items will be addressed in a supplement to this Safety Evaluation Report. These items are listed below and are discussed further in the sections of this report as indicated.

(1) We require the applicant to conform to the recommendations of the Commission's Regulatory Guides 1.8, 1.63, 1.67, and 1.79 or for each guide provide an acceptable alternative solution to the safety matter addressed by the guide (Section 1.11.4).

(2) We require additional information on foundations for above-ground water storage tanks and for buried diesel generator fuel oil tanks (2.5.3(4)).

(3) We require that the chimney drain in the nuclear service water dam be increased from a width of three feet to a width of six feet and that a twelve-foot wide impervious embankment zone be provided upstream and adjacent to the chimney drain and to the blanket drain in the core trench (2.5.3(7)).

(4) We require that the applicant provide criteria for assuring that postulated failures of non-seismic Category I structures would not impair the capability of seismic Category I structures to perform their design function (Section 3.7.2).

(5) We require a commitment by the applicant to provide acceptable equipment design modifications in the FSAR to preclude water-solid overpressurization of the reactor coolant system (Section 5.2.2).

(6) We require that the applicant provide sufficient justification for excluding several paths from his proposed list of containment leakage paths that bypass the secondary containment (Section 6.2.4). We will require a resolution of this matter prior to completion of our review.

(7) Unless new bases are developed which we find acceptable, we require that periodic local containment leakage rate testing be accomplished without use of water as a pressurizing medium (Section 6.2.4).

(8) For the fuel handling accident, the applicant has referenced Section 15.4.6.1 of the CESSAR which states "Release of activity through the Containment Purge Systems would be prevented by automatic closure of the containment isolation dampers. The containment personnel and equipment hatches are closed during fuel handling operations. Since the Auxiliary Building cannot be completely isolated,

this results in a more limiting activity release to the environment." Consistent with this statement, we calculated the offsite doses that would result from a postulated fuel handling accident in the fuel handling portion of the auxiliary building (Section 15.5.6 of this report). However, as noted in Section 6.2.4 of this report, the applicant has not described the provisions for containment isolation required by the CESSAR reference in order that the above statement be valid for the proposed facility. We require a resolution of this matter by the applicant prior to the completion of our review.

(9) We require the applicant to document a commitment that suitable design modifications will be provided if his analysis shows that the consequences of two steam generators' blowing down are not acceptable (Section 7.1.2 of this report).

(10) Evaluation of the applicant's exceptions to the Commission's Regulatory Guides 1.14, 1.31, 1.44, 1.50, (Section 1.11.5).

(11) Evaluation of the applicant's proposed exceptions to CESSAR interface requirements on use of delta ferrite in austenitic stainless steel and on water quality of the emergency feedwater (Section 1.13 items 1 and 6).

(12) Evaluation of the applicant's financial qualification (Section 20.0).

1.10 Generic Issues

The Advisory Committee on Reactor Safeguards periodically issues a report listing various generic matters applicable to light water reactors. Our discussion of these matters is provided in Appendix C to this report which includes references to sections of this report for more specific discussions that particularize for the proposed facility the generic status.

In addition to the generic matters identified by the Committee that are listed in Appendix C, the following are matters for which the staff is conducting generic reviews that may impact the design of the proposed facility.

(1) Design provisions to preclude water-solid overpressurization of the primary coolant system (Section 5.2.2).

(2) Evaluation of fuel rod bowing effects (Section 15.4).

(3) Anticipated transients without scram (Section 15.6 and Section 15.6 of Appendix A).

1.11 Exceptions to Regulatory Guides

The applicant states in Section 1.7 of the PSAR that he reviews each Regulatory Guide generally within six months of issuance of the guide. Table 1.7-1 of the PSAR, which is a summary of the applicant's position on each Regulatory Guide, was updated to November, 1976 by Amendment 28 to the PSAR. By letter of February 8, 1977, the applicant stated that Table 1.7-1 will be updated in Amendment 29. That updating as described in the February 8, 1977 letter has been reflected in our review.

We did not review the proposed exceptions to five of the Regulatory Guides (Section 1.11.1 below), determined that exceptions to two of the Regulatory Guides were not applicable to the proposed facility (Section 1.11.2), found exceptions to seven of the Regulatory Guides were acceptable for the proposed facility (Section 1.11.3), found exceptions to four of the Regulatory Guides were not acceptable (Section 1.11.4), and have not completed our review of proposed exceptions to five Regulatory Guides (1.11.5).

1.11.1 Exceptions to Regulatory Guides Not Relied on in the Review

We did not review the proposed exceptions to five Regulatory Guides because we used the recommendations of the guides for our independent analyses and did not rely on the applicant's information in the areas of his proposed exceptions. The five guides for which we did not review the proposed exceptions are:

Regulatory Guide 1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Pressurized Reactors (Rev. 2)" (Sections 2.3.4 and 15.5.6).

Regulatory Guide 1.24 "Assumptions used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure (Rev. 0)" (Section 15.5.6).

Regulatory Guide 1.25 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Rev. 0)" (Section 15.5.6).

Regulatory Guide 1.77 "Assumptions Used for Evaluating a Control Rod Ejection Accident for PWR's (Rev. 0)." We completed our review of the CESSAR using the methods of Regulatory Guide 1.77 to perform an analysis of the radiological consequences of a postulated control rod ejection accident (Section 15.5.6 of Appendix A). For our short term diffussion estimates for the proposed site (Section 2.3.4), the results of our analysis in Appendix A show a need for a reduction in the primary to secondary leak rate below the one gallon per minute assumed in Appendix A (Section 15.5.6) in order to maintain calculated offsite doses within the guideline values of Appendix A. Since the analysis methods of Regulatory Guide 1.77 are conservative, a future finding by the staff that the CESSAR exceptions are acceptable would likely permit technical specifications at the operating license stage of review for less restrictive primary to secondary leak rates.

Regulatory Guide 1.111 "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors (Rev. 0)." (Section 2.3.5)

1.11.2 Exceptions to Regulatory Guides Not Applicable

We reviewed and concluded that proposed exceptions to two Regulatory Guides were not applicable because of the design features of the proposed facility. (We did not review the applicability to other facilities licensed or proposed by the applicant.) The two Regulatory Guides with proposed exceptions that are not applicable to the proposed facility are:

Regulatory Guide 1.78 "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release" (Section 6.5.2).

Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release (Rev 0)" (Section 6.5.2).

1.11.3 Acceptable Exceptions to Regulatory Guides

We have reviewed and found acceptable for the proposed facility exceptions proposed by the applicant to the following seven Regulatory Guides:

Regulatory Guide 1.46 "Protection Against Pipe Whip Inside Containment (Rev. 0)." As reported in Section 3.6 of Appendix A to the report, we find that the CESSAR is consistent with Regulatory Guide 1.46. For systems inside the containment but not within the scope of the CESSAR, the applicant proposed exceptions to Regulatory Guide 1.46 in Table 3.6.1-3 of the PSAR. We reviewed these exceptions and found them acceptable for the proposed facility (Section 3.6.1).

Regulatory Guide 1.52 "Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (Rev. 1)." The applicant states that the various air filtration systems comply as applicable with the positions of Regulatory Guide 1.52 and cites Table 6.2.3.3 of the PSAR for clarification and applicability (PSAR Section 6.2.3). We have completed our review and conclude that the extent of conformance with the provisions of Regulatory Guide 1.52 is acceptable (Section 6.2.8).

Regulatory Guide 1.54 "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants (Rev. 0)." The applicant states that the materials and components will be compatible with both the normal operating environment and the most severe thermal, chemical and radiation environment expected during post-accident conditions and that surface preparation and coatings compatible with the exposure and environment will be in accordance with Table 6.2.1-16 of the PSAR (PSAR Section 6.2.1.6). We have reviewed this information and find the applicant's provisions for quality assurance for protective coatings to be acceptable (Section 17.2 staff review of Duke Power Company Topical Report, "Quality Assurance Program-Duke 1").

Regulatory Guide 1.58 "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (Rev. 0)." We found the applicant's exceptions to be acceptable during our review of Duke Power Company's QA program described in their topical report on quality assurance, "Quality Assurance Program-Duke 1" as modified by Amendments 1, 2 and 3 (Section 17.2).

Regulatory Guide 1.75 "Physical Independence of Electric Systems (Rev. 1)." The applicant has provided clarifying statements pertaining to his conformance to the provisions of Regulatory Guide 1.75 (PSAR Section 8.3.1.2.6). We have reviewed this information and find acceptable conformance with Regulatory Guide 1.75.

Regulatory Guide 1.80 "Preoperational Testing of Instrument Air Systems (Rev. 0)." By letter of February 8, 1977, the applicant committed to performing preoperational tests in accordance with the provisions of Regulatory Guide 1.80 except for tests C9 and C10 specified by the guide. The applicant claims test C9 is not necessary because, for the proposed design, moisture freezing in the lines is not credible and test C10 would duplicate test C8. We have reviewed this matter and conclude that the applicant's exceptions are acceptable for the proposed facility design.

Regulatory Guide 1.89. "Qualification of Class IE Equipment for Nuclear Power Plants (Rev. 0)." The applicant states that exceptions to the requirements of IEEE 323-1974 will be stated in the implementation program and justified (PSAR Section 3.11.2.1.) However, in Section 3.11, we conclude the applicant's commitments to meet the requirements of IEEE 323-1974 are acceptable on this basis. On this basis we conclude there will be no exception to Regulatory Guide 1.89 having a safety significance.

1.11.4 Non-Acceptable Exceptions to Regulatory Guides

We have completed our review and find exceptions proposed to four Regulatory Guides to be unacceptable to us on the basis of information presented by the applicant. We will require that the applicant commit to conform to the recommendations of each these guides or provide an alternative solution to the safety matter addressed by the guide that provides a level of safety equivalent to that of the guide. These four Regulatory Guides are:

Regulatory Guide 1.8 "Personnel Selection and Training." We will require conformance with our position that acceptable qualifications for the position of Radiation Protection Manager include nine years' experience (Section 13.1).

Regulatory Guide 1.63 "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants (Rev. 0)." The applicant proposes that those circuits which are incapable of supplying a fault current sufficient to cause loss of mechanical integrity of the penetration will not be provided with overload protection

712 033

(PSAR Section 8.3.1.2.11). We agree with the applicant that for thermocouple instrumentation circuits and computer points overload protection is not needed. However, for the annunciator circuits, we will require a commitment that an analysis be provided in the FSAR to demonstrate to our satisfaction that protection is not needed, or that failing such demonstration, protection will be provided.

Regulatory Guide 1.67 "Installation of Overpressure Protection Devices (Rev. 0)." By letter of February 8. 1977, the applicant proposes an exception to American Society of Mechanical Engineers (ASME) Code Case 1569, which is incorporated in the provisions of Regulatory Guide 1.67. The proposed exception is to use nominal inlet pipe size instead of nominal discharge pipe size in computing the "branch moment." On the basis of our review, and in the absence of favorable action by an ASME Code Committee we do not find the proposed exception acceptable. We will require that the applicant withdraw this exception.

Regulatory Guide 1.79 "Preoperational Testing of ECCS for PWR's (Rev. 1)." The applicant in Table 1.7-1 references Section 3.1.33 of the CESSAR which in turn shows no exception to Regulatory Guide 1.79. Since this is a preoperational test we will require a confirmation of this commitment by the applicant in the PSAR.

1.11.5 Exceptions to Regulatory Guides-Review Not Completed

We have not completed our review of exceptions proposed to five Regulatory Guides. We will report the results of our review in a supplement to this report. The five Regulatory Guides are:

Regulatory Guide 1.14 "Reactor Coolant Pump Flywheel Integrity (Rev. 1)." We conclude in Appendix A to this report that the conformance to the recommendations of Regulatory Guide 1.14 committed to in the CESSAR constituted an acceptable basis for satisfying the requirements of Criterion 4 of the Commission's General Design Criteria. In Amendment 28 the applicant in Table 1.7.1 of the PSAR has committed to partial compliance with the in-service inspection recommendations of Regulatory Guide 1.14.

Regulatory Guide 1.31 "Control of Stainless Steel Welding (Rev. 1)." The applicant has proposed an extensive list of requirements in lieu of the provisions of Regulatory Guide 1.31 for control of stainless steel welding (PSAR Section 5.2.5.7).

Regulatory Guide 1.44 "Control of the Use of Sensitized Stainless Steel (Rev. 0)." The applicant has proposed an extensive list of requirements in lieu of the provisions of Regulatory Guide 1.44 for control of the use of stainless steel (PSAR Section 5.3.5.8).

Regulatory Guide 1.50 "Control of Preheat Temperature for Welding of Low-Alloy Steel (Rev. 0)." The applicant has taken an exception to paragraph 6.2 of Regulatory Guide 1.50 pertaining to the maintenance of preheat until stress relief is performed (PSAR Section 5.2.3.5).

Regulatory Guide 1.71 "Welder Qualification for Areas of Limited Accessibility (Rev. 0)." The applicant takes exception to the access provisions of the guide (PSAR Section 5.2.3.9).

1.12 Exceptions to the CESSAR

In Section 1.9.3.3 of the PSAR the applicant states that the PSAR incorporates by reference the major portions of the CESSAR, and that, however some subsections of CESSAR are not incorporated in their entirety. The applicant states that this situation is due to contractural options with the nuclear steam supply system vendor which the applicant has exercised and the use of alternative design criteria or code requirements. In Amendment 29 to the PSAR, the applicant will withdraw the following two exceptions listed in Section 1.9.3.3 of the PSAR.

3.10 Seismic Design of Class IE Instrumentation and Electrical Equipment.

10.4.4 Turbine Bypass System (Footnote in Table 1.2-1, sheet 2 of 2 is applicable).

Section II B, "Post Review Aspects" of WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants" strongly discourages changes to the standard design as proposed by the applicant. Nevertheless, we have reviewed the applicant's four remaining proposed exceptions to determine whether they would invalidate our reliance on our review of the CESSAR as described in Appendix A to this report.

The results of our review of the four proposed exceptions are as follows:

3.11 Environmental Design of Mechanical and Electrical Equipment. We will require that mechanical and electrical equipment within the scope of the CESSAR, and located in the containment be qualified to containment environmental conditions at least as severe as those for which the mechanical and electrical equipment within the scope of the PSAR will be qualified (Section 3.11).

5.2.5.7 Percentage of Delta Ferrite Used in Austenitic Stainless Steel. We will require that for stainless steel within the scope of the CESSAR, the applicant include the same percentage of delta ferrite as specified in the CESSAR. We have not reviewed the CESSAR for the applicant's proposed exception to this percentage of delta ferrite.

4.0 Reactor. The applicant proposes that eight additional control element assemblies (CEA's) be installed in the locations designated as spares on CESSAR Figure 4.2-19. We did not complete a review of such a proposed configuration (Appendix A to this report), and the applicant did not provide a safety analysis. [illegible], we cannot find this proposed exception acceptable on the basis of information now available.

9.1.4.2 Spent Fuel Handling Machine. The applicant states that he has elected to use a spent fuel handling machine that is identical to the refueling machine described in the CESSAR. We have concluded that the use of the CESSAR refueling machine design for the spent fuel handling machine in the proposed facility instead of the CESSAR spent fuel handling machine design is acceptable (Section 9.1.4).

Unless specifically excluded by this report, any equipment within the scope of the CESSAR is subject to the provisions of the CESSAR independent of the source of supply, including the applicant as a supplier.

1.13 Exceptions to CESSAR Interface Requirements

The applicant in Section 1.9.3.3 of the PSAR has identified, and characterizes as minor, variations to the CESSAR interface requirements that he proposes. The results of our review are as follows:

(1) Delta ferrite - The exception is that expressed as an exception to Regulatory Guide 1.31 (Section 1.12 of this report). We will report the results of our review in a supplement to this report.

(2) Source terms - The applicant has proposed source terms for the liquid and gaseous radioactive waste management systems and for release to containment following a postulated loss-of-coolant accident. We have used source terms that we normally use in our review and did not rely on the terms proposed by the applicant (Sections 11.2, 11.3 and 15.5.6).

(3) Environmental Design Conditions - We have made our independent calculations of environmental conditions within the containment and conclude that applicant's conditions proposed for qualification of mechanical and electrical equipment will be acceptable (See Section 3.11).

(4) Water Pumped Gases - The interface requirement is for use of water in pumping gases. The applicant has committed to minimum purity requirements of the CESSAR without yet choosing the compression process. We conclude that this commitment is acceptable for the construction permit stage of review.

(5) Instrument Accuracy - The applicant proposes to increase the inaccuracy of feedwater temperature measurement from plus or minus one degree Fahrenheit as required by a CESSAR interface requirement to plus or minus two degrees Fahrenheit. This is acceptable for the construction permit stage of review, but appropriate justification will be required at the operating license stage of review to show that performance analyses are applicable for this instrument capability.

(6) Water Quality - A CESSAR interface requirement is that the quality of emergency feedwater shall be that of normal reactor coolant makeup water. The applicant proposes the same quality as for the condensate in the exterior condensate storage tank. We will complete the results of our review of this matter in a supplement to this report.

(7) Electrical Instrumentation - The CESSAR includes interface requirements that require sources of electric power to the reactor protective system and engineered safety features actuation system be ungrounded. The applicant proposes that the direct current sources remain ungrounded and that the alternating current sources be grounded. We conclude that including provisions for grounding of the alternating current sources is compatible with the CESSAR systems and is acceptable.

A CESSAR interface requirement is that CO-AX or TRI-AX cable used for nuclear instrumentation shall be run its entire length in conduit. The applicant's exception proposes the use of a small separate cable tray in lieu of conduit, which we concluded is acceptable.

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Description

The site is located in the eastern portion of Cherokee County, South Carolina, approximately 40 miles southwest of Charlotte, North Carolina, and 21 miles east of Spartanburg, South Carolina. The geographic location of the site is shown in Figure 2.1. The site property is bordered on the north and east by the Broad River, on the south by South Carolina Highway 13 or by property fronting on Highway 13, and on the west by private property and is directly west of Duke Power Company's Ninety-Nine Islands Hydroelectric Station. The locations of the proposed facility features relative to the locations of existing features are shown in Figure 2.2.

2.1.2 Exclusion Area Control

The exclusion area for the site is a 2500-foot (762 meters) diameter circular area extending from the center of the middle containment of the three containments. The minimum exclusion area distance for the two end units is 1960 feet (594 meters). Duke Power Company owns all land within the exclusion area. We conclude that this ownership constitutes adequate assurance that the applicant can provide control of the land within the exclusion area in accordance with the requirements of Section 100.3 of 10 CFR Part 100. The exclusion area includes parts of the Ninety-Nine Islands Reservoir and the Broad River. Prior to a decision to issue construction permits, the applicant must demonstrate that it can control the movement of persons in these water areas in the event of an emergency.

2.1.3 Population and Population Distribution

In Figures 2.1.3-3 through 2.1.3-14 of the PSAR, the applicant has shown population data or population projections out to a distance of 50 miles from the site for the years 1970, 1980, 1990, 2000, 2010 and 2020. The data for the year 1980 shown in Figures 2.1.3-4 and 2.1.3-10 of the PSAR and shown in Figure 2.3 of this report as cumulative population illustrate that the population at all distances out to 50 miles from the site is less than 500 people per square mile, which is a population density that we use to characterize a moderately populated area.

We obtained an independent estimate of the 1970 population within 50 miles of the site from Bureau of the Census data and found that our population figures agreed reasonably well with the applicant's value of 1,308,327. The applicant's projected population growth rate to the year 2020 for the area within 50 miles of the site was compared to the population projections of the Bureau of Economic Analysis for Economic

712 038

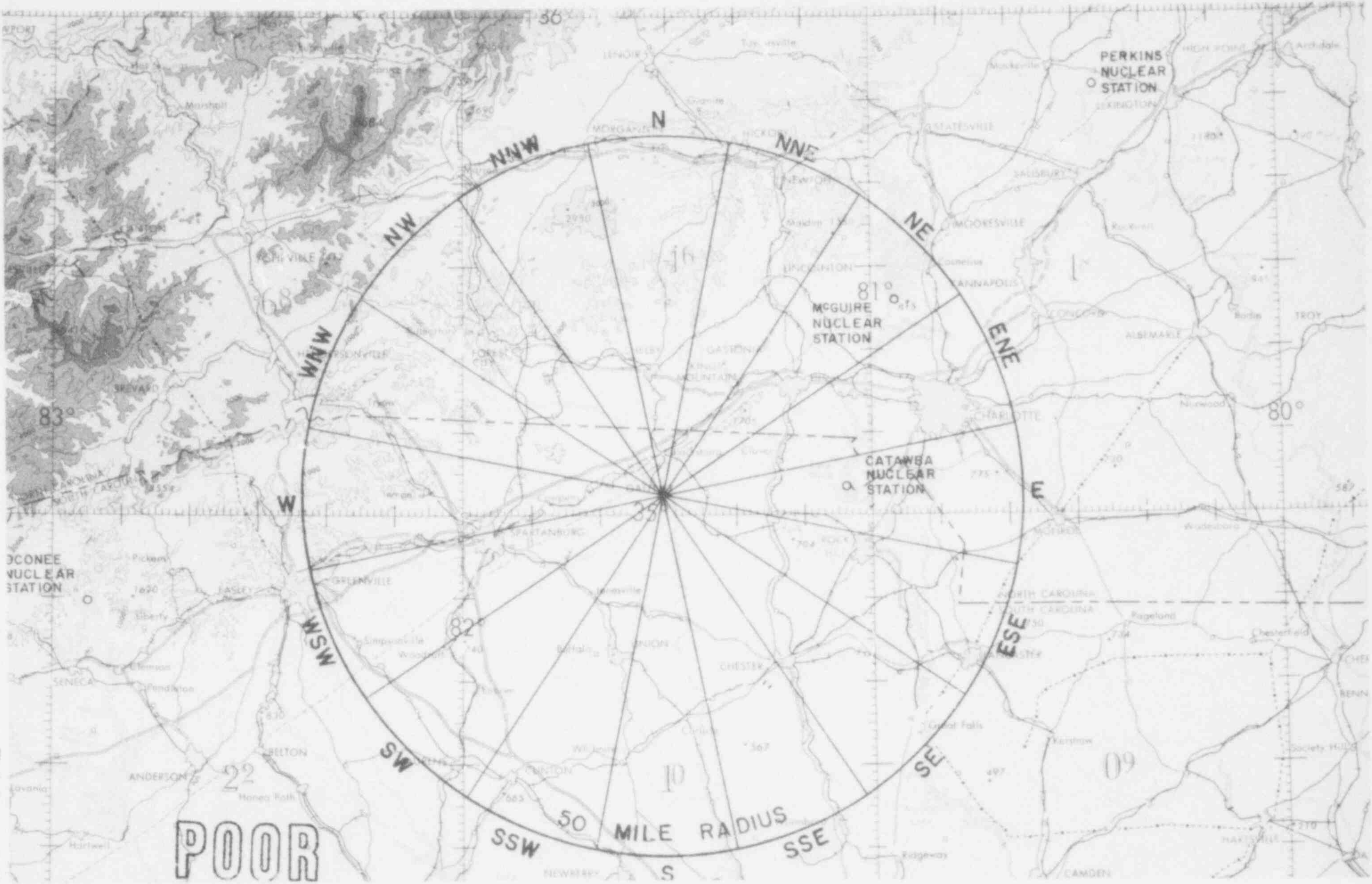


Figure 2.1 General Area Map

Figure 2.2 Plot Plan and Site Boundary

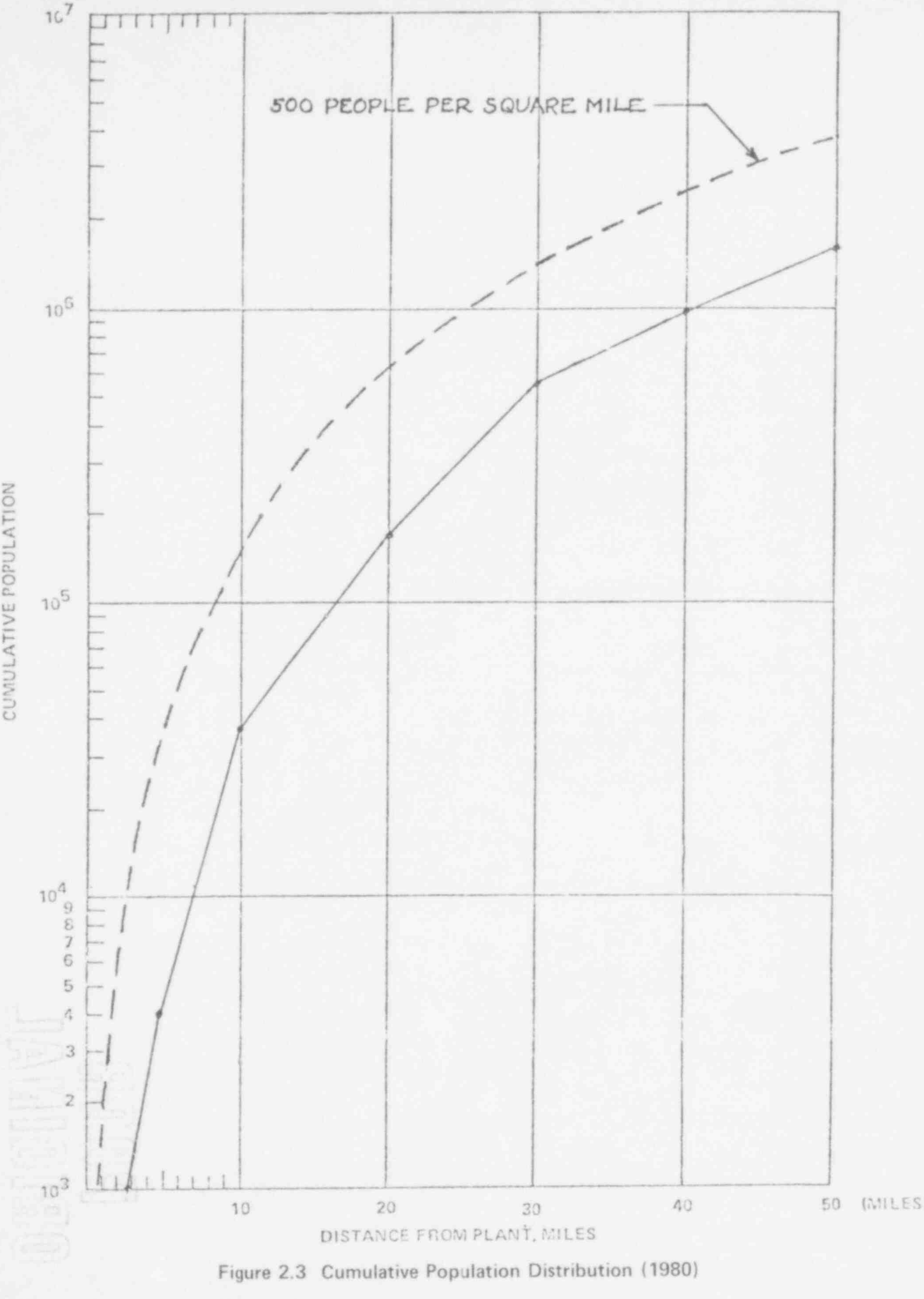


Figure 2.3 Cumulative Population Distribution (1980)

Area 28, which is shown in Figure 2.4. This comparison showed that the applicant's growth projections of about 16 percent per decade, for the 50-mile area was higher than the Bureau of Economic Analysis projection of 10 percent per decade for Area 28.

The applicant has selected a low population zone with an outer radius of five miles. The total 1970 resident population within the low population zone was about 3500 persons. Transient population within the Cherokee low population zone is estimated to include 250 at Burlington Industries, 2.5 miles northeast of the site, up to 35 per day at the reservoir near the Ninety-Nine Islands Dam, and 30 per day at local game management areas; transient population outside the low population zone includes up to 1300 per day at Kings Mountain National Military Park eight miles northeast of the site.

As a result of our evaluation of the applicant's proposed low population zone, we conclude that there is reasonable assurance that the 10 CFR Part 100 definition of the low population zone can be satisfied in that we have not identified any unusual characteristics with respect to the low population zone which would prevent the development of appropriate emergency response procedures (see Section 13.3 of this report).

The population center nearest the Cherokee site, as defined in 10 CFR Part 100, is Spartanburg, South Carolina, which is about 21 miles from the site and which had a 1970 population of about 45,000 persons. Gaffney, South Carolina, had a population of about 13,250 in 1970. Our projections suggest that it is unlikely that Gaffney would become a population center until very late in the plant lifetime. Since Gaffney is about eight miles from the site, a need for a major reduction in the low population zone distance during the plant lifetime would not be necessary even if a population center were to develop at Gaffney. As is indicated in Table 15.1 of this report, the calculated low population zone 30-day doses are a small fraction of the guideline doses of 10 CFR Part 100. Thus, major reductions in the low population zone distance would be possible for the proposed facilities. The present population center distance of 21 miles is well in excess of the minimum distance of one and one-third times the low population zone distance required by 10 CFR Part 100. These requirements can also be satisfied for population centers as close or even closer than a potential population center at Gaffney without any changes in engineered safety features for the facility.

2.1.4 Conclusion

On the basis of the 10 CFR Part 100 definitions of the population center distance the exclusion area distance, and the low population zone radius, our estimate of the 1980 population distribution, our analysis of the onsite meteorological data from which relative concentration factors were calculated (See Section 2.3 of this report), and the calculated potential radiological dose consequences of design basis accidents (See Section 15.0 of this report), we have concluded that the exclusion area, low population zone and population center distance meet t siting criteria of 10 CFR Part

712 042

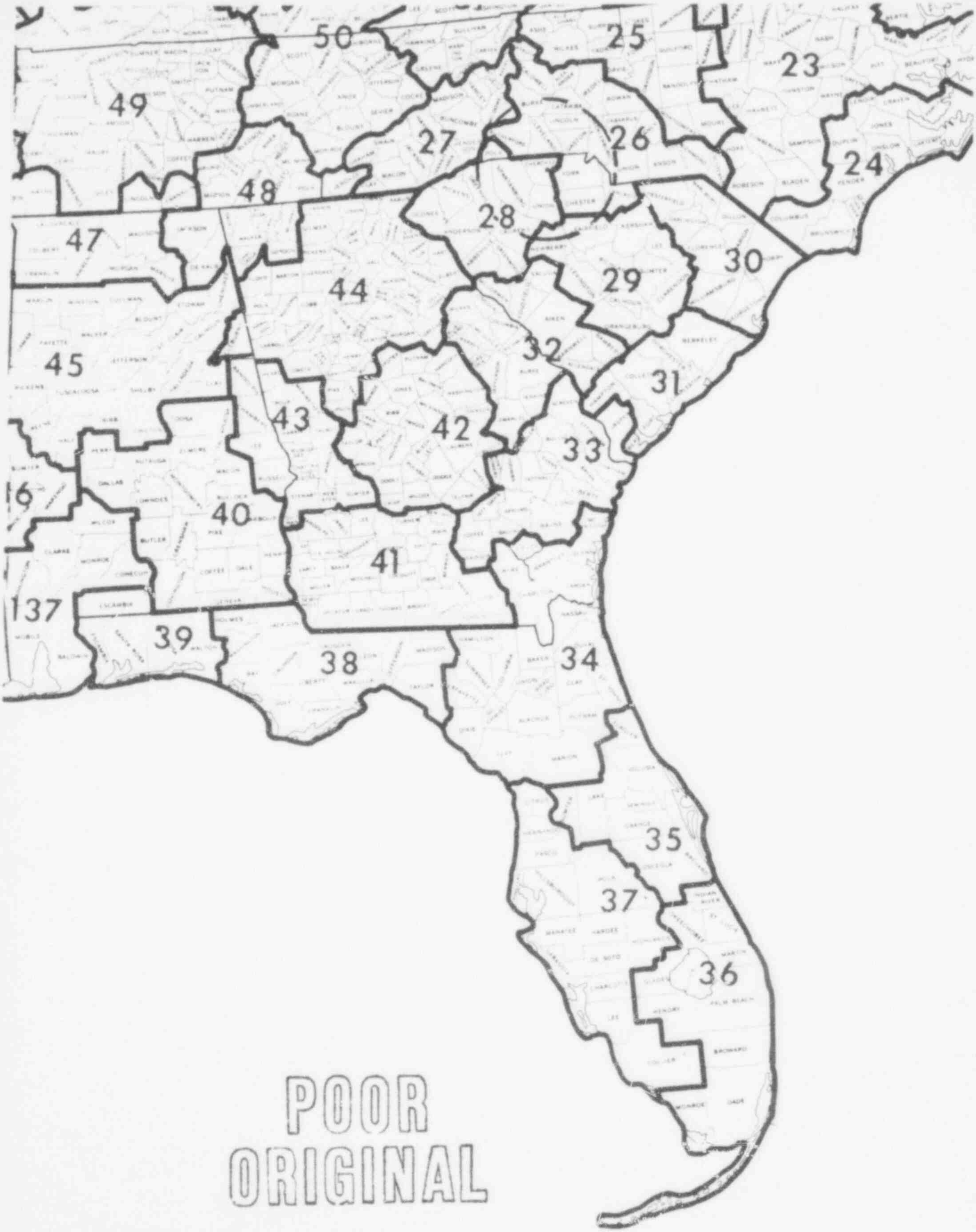


Figure 2.4 BEA Area No. 28 With Plant Site Location and 50-Mile Radius Around Cherokee Nuclear Station

100 and are acceptable. We further conclude that these siting criteria can be met for all population centers that we have projected as potential population centers that could develop during the plant life for the proposed facility.

2.2 Nearby Industrial, Transportation and Military Facilities

Nearby industrial and transportation facilities are shown in Figure 2.5. The nearest industry to the proposed facility is Burlington Industries, a manufacturer of cotton and gray goods, located two and one-half miles northwest of the site. We find that this activity will pose no potential hazard to the plant. A pipeline corridor approximately four miles northwest of the site includes pipelines that carry refined liquid petroleum products and pipelines that carry methane gas. We have investigated the hazards associated with these pipelines and have concluded that the pipelines do not pose a significant threat to plant safety. There are no other industrial facilities within five miles of the plant location.

There are presently no State or U.S. highways within four miles of the site. The nearest airport is the Cherokee Airport with a sod runway located nine miles west-northwest of the site. There is a railroad line of Southern Railways five to six miles from the site which we conclude would pose no hazard to the proposed Cherokee facility.

We have investigated the federal and military airways identified in Section 2.2.1 of the PSAR and conclude that activities on these airways pose no threat to safe plant operation.

On the basis of our evaluation, we conclude that nearby industrial, transportation, and military facilities pose no threat to safe plant operation.

2.3 Meteorology

2.3.1 Regional Climatology

The climate of the Cherokee site, located about 40 miles west-southwest of Charlotte, North Carolina, is typical of continental climates in southern areas and is characterized by cool winters and relatively long, warm summers. Cold air moving southward into the area is modified somewhat by crossing the Appalachian Mountains.

The site may be affected by thunderstorms, tornadoes, tropical storms and hurricanes.

Thunderstorms can be expected to occur on about 42 days per year, with the period May through August having 30 thunderstorm days. Hail greater than three-fourths of an inch in diameter within the 13-year period 1955-1967 has been reported 14 times in the region of the site. Also in this period, 20 windstorms (excluding tornadoes) with gusts greater than 50 knots (58 mph) occurred within the one-degree latitude-longitude square that contains the site.

Figure 2.5 Nearby Industrial and Transportation Facilities

712 045

In the period 1871-1974, 27 tropical depressions, storms, and hurricanes passed within 50 miles of the site.

During the period 1955-1974, four tornadoes were reported in the one-degree latitude-longitude square containing the site, giving a mean annual frequency of 0.3 per year. The computed recurrence interval for a tornado at the plant site is 4400 years. The design basis tornado characteristics selected by the applicant for the site conform to the recommendations of Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants" (April 1974), for this region of the country.

The "fastest mile" wind speed recorded at Charlotte was 74 miles per hour. The operating basis wind speed (defined as the "fastest mile" wind speed at a height of 30 feet above the ground with a return period of 100 years) of 95 miles per hour selected by the applicant is adequate for the proposed site.

The applicant has examined meteorological data from Charlotte, North Carolina, for the period 1949 through 1973 to select the appropriate design basis meteorological conditions to be considered in the design of the ultimate heat sink as recommended in Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants" (Revision 1, March 1974). We find the meteorological data presented in Table 9.2.5-4 of the PSAR for analyses of minimum water cooling and the data presented in Table 9.2.5-5 of the PSAR for analyses of maximum water loss reasonably conservative, and are acceptable for design of the ultimate heat sink.

In the period 1936-1970, there were about 84 atmospheric stagnation cases totaling about 325 days reported in the area of the site. The maximum monthly frequency occurs in October.

2.3.2 Local Meteorology

Climatological data from Charlotte and from Greenville-Spartanburg Airport (about 40 miles west of the site), and available onsite data have been used to assess local meteorological characteristics of the site.

Mean monthly temperatures at the site may be expected to range from about 42 degrees Fahrenheit in January to about 79 degrees Fahrenheit in July. Record extreme temperatures in the site area have been 104 degrees Fahrenheit and six degrees Fahrenheit.

Annual average precipitation in the site area is about 43 inches. The maximum mean monthly precipitation of about 4.6 inches occurs in July, while the minimum mean monthly precipitation of about 2.7 inches occurs in October and November. Annual average snowfall is about five inches.

Wind data from the 30-foot level at the site for the period September 11, 1973, through September 10, 1974, indicate prevailing wind directions from the southwest and the northwest directions with frequencies of 11 percent for each direction.

These predominant wind directions evidently reflect drainage flow patterns under certain synoptic conditions. Winds from the south-southeast occurred least frequently with a frequency of about two percent. Calms occurred about 5.5 percent of the time. The average wind speed at the 30-foot level for this time period was 3.6 miles per hour.

2.3.3 Onsite Meteorological Measurements Program

The preoperational onsite meteorological program was initiated in September 1973. A 30-foot tower and a 130-foot tower (a converted electrical transmission tower) were located on the site at the proposed location of the cooling towers. As stated in Amendment 13 to the PSAR (page 2.3-10), these towers will be replaced by a permanent meteorological facility to become operational "as soon as practical after site excavation has rendered the elevations and exposure representative of final plant conditions."

Wind speed and direction were measured at the top of the 30-foot tower. On the 130-foot tower, vertical temperature difference was measured between 30 feet and 130 feet, tower wind speed and direction were measured at the 130-foot level, ambient air and dewpoint temperatures were measured at 30 feet and precipitation was measured near the ground. The data were recorded on strip charts.

The accuracy capability of the vertical temperature difference data system did not initially conform to the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs", however, the applicant, in November 1974, installed instruments that conform to the accuracy recommendation, and performed a comparison of relative concentration values using data from both systems for the same month. Our independent analysis of these data indicate that relative concentration values calculated differ by only about ten percent. Therefore, we conclude that the meteorological measurements program meets the recommendations and intent of Regulatory Guide 1.23.

The applicant has submitted one full year (September 11, 1973 through September 10, 1974) of onsite joint frequency distributions of wind speed and direction at the 30-foot level by atmospheric stability (as defined by the vertical temperature gradient between 30 feet and 130 feet) in the format suggested in Regulatory Guide 1.23. The data recovery rate was 93 percent. Similar distributions were submitted with wind data from the 130-foot level of the onsite tower, with a data recovery rate of 92 percent. Also submitted were joint frequency distributions (with stability defined by the STAR program) for a five-year period (1968-1972) from Greenville-Spartanburg Airport.

We have examined relative concentration values using each joint frequency distribution. Relative concentration values calculated using each distribution were not significantly different in magnitude for pertinent distances and directions.

The relative concentration values presented in Sections 2.3.4 and 2.3.5 are based on the onsite data with wind speed and direction measured at 30 feet.

712 047

2.3.4 Short-Term (Accident) Diffusion Estimates

Using Duke Power Company's onsite meteorological data and the diffusion model described in Regulatory Guide 1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Reactors," we have made conservative assessments of atmospheric diffusion concentration values for the various time periods following a postulated accidental release. In our evaluation of short-term (0-2 hours at the exclusion distance and 0-8 hours at the low population zone distance) accidental releases from the plant buildings and vents, we assumed a ground-level release with the applicant's building wake factor, cA, of 808 square meters.

The relative concentration for the 0-2-hour time period which is exceeded no more than five percent of the time is 2.5×10^{-3} seconds per cubic meter at the exclusion distance of 594 meters. This relative concentration is equivalent to that calculated assuming Pasquill Type F stability with a wind speed of 0.3 meters per second.

The relative concentration values for various time periods at the outer boundary of the low population zone (8000 meters) are:

Time Periods	Relative Concentrations in Seconds Per Cubic Meter
0-8 hours	5.9×10^{-5}
8-24 hours	3.8×10^{-5}
1-4 days	1.5×10^{-5}
4-30 days	4.0×10^{-6}

2.3.5 Long-Term (Routine) Diffusion Estimates

Using the diffusion model described in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," and Duke Power Company's onsite meteorological data, we calculated the highest offsite annual average relative concentration for vent releases assuming a ground-level release to be 2.4×10^{-5} seconds per cubic meter at the site boundary (594 meters) east of the proposed reactor complex.

2.3.6 Conclusions

We have concluded that the meteorological data presented by the applicant for the period from September 11, 1973 to September 10, 1974 provide an acceptable basis for determining conservative estimates of atmospheric dispersion to be used calculating accidental and routine gaseous releases from the Cherokee facility.

712 048

2.4 Hydrology Engineering

2.4.1 Hydrologic Description

The Cherokee site is located on the west bank of the Broad River approximately 91 miles upstream of its confluence with the Saluda River and just upstream of the applicant's Ninety-Nine Islands Dam and Hydro Station. McGowan Creek, to the west, will be impounded to form a reservoir which will serve as the nuclear service water pond. To the east of the site is a storage impoundment for the Ninety-Nine Islands (run-of-the-river) Hydroelectric Station. The intake sedimentation basin which will be formed by impounding a leg of Ninety-Nine Islands impoundment will provide holdup of water pumped from the Broad River to remove sediment from the water prior to its use as plant makeup water. The river intake structure will be located on the west bank of the Broad River about 1000 feet upstream of the Ninety-Nine Islands Dam. Other major hydraulic structures at the site will include (1) reservoir embankments and associated water control structures for the nuclear service water pond and the alternate nuclear service water pond, (2) pump houses for the plant water systems, and (3) the seismic Category I permanent underdrain system and exterior wall drains.

The Broad River begins in the eastern foothills of the mountains in western North Carolina and flows in a southeasterly direction to a point near Gaffney, South Carolina. It then flows in a southerly direction to Columbia, South Carolina, where it is joined by the Saluda River to form the Congaree River. The Congaree River joins the Wateree River near Eastover, South Carolina, forming the Santee River which flows southeasterly into the Atlantic Ocean near Georgetown, South Carolina. The Broad River has a length of approximately 185 miles and a drainage area of about 5,240 square miles. The drainage area at the site is about 1550 square miles. The river is gauged at the Gaffney gauge about five miles upstream of the site where the drainage area equals 1490 square miles. The mean annual flow at the gauge is about 2470 cubic feet per second. The maximum and minimum flows of record are 119,000 cubic feet per second and 140 cubic feet per second, respectively. The Broad River is generally wide and shallow and carries a large sediment load, including a large bedload comprised mostly of sand. There are several upstream reservoirs that partially control the flow at the site. Lake Lure, Lake Adger and Lake Summit are headwater reservoirs, located in the upper reaches of the Broad River and its tributaries. The nearest major upstream river control structure is the Gaston Shoals Dam, which is a run-of-the-river hydroelectric plant located about 10 miles upstream of the site.

The Ninety-Nine Islands Hydro Station and dam were completed in 1910 by the applicant. The original surface area and volume of the reservoir were 885 acres and 21,240 acre-feet at the full pond elevation of 511 feet above mean sea level. The pond is now heavily silted with a surface area of about 350 acres and a normal pool volume of 2260 acre-feet.

2.4.2 Flooding Potential

Several potential flood-producing sources including the Broad River, site drainage in the vicinity of safety-related structures, potential dam failure, ice flooding, surges, and tsunamis, were investigated by the applicant and evaluated by the staff as follows:

(1) Historically, the flood flows in the Broad River have been caused by tropical storms that moved ashore and inland, usually in the summer and early autumn months. The August 14, 1940 flood was the maximum flood of record listed in the U.S. Geological Survey Water Supply Papers for the Gaffney gauge. This flood had a maximum computed flow over the Gaston Shoals Dam of about 119,000 cubic feet per second. This flow corresponds to an estimated maximum stage, at the proposed location of the facility river intake structure, of about elevation 522.0 feet above mean sea level. The applicant has quoted the maximum flood of record for the Ninety-Nine Islands drainage areas as the flood of July 1916. This flood had a discharge of about 133,000 cubic feet per second and a static water level at the proposed location of the facility river intake of 524.1 feet above mean sea level. The computed maximum elevation, including wind tide and wave runup resulting from a 40 mile per hour overland wind, would be 527.7 feet mean sea level. The source of this historic flood data was not given, although presumably it is from unpublished records for the Ninety-Nine Islands Hydro Station.

Three criteria used as bases for design flood levels on the Broad River were (a) the probable maximum flood due to probable maximum precipitation, (b) the complete failure of any upstream dam(s) during the probable maximum flood, and (c) the seismic failure of any upstream dam(s) coincident with the standard project flood. The highest water surface elevation calculated for the above conditions is elevation 567.4 feet above mean sea level for criterion (b), which assumes a surge wave from the domino failure of Tuxedo and Turner Dams and the failure of Lake Lure Dam, such that these peaks would coincide with the probable maximum flood peak stage at the plant site. We concur that this is a conservative analysis based upon our recommended design basis flood criteria described in Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," August 1973, and with the resultant peak stage of 567.4 feet above mean sea level as the probable maximum flood level at the site. This peak stage is 22.6 feet below the plant grade elevation of 590 feet above mean sea level, and we conclude that the site will not be flooded from any reasonably possible combination of probable maximum flood and upstream dam failures on the Broad River.

(2) We conclude that the proposed facility river intake structure is not a necessary feature for the facility design to be in accordance with the provisions of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants" and that our review of the design considerations is not necessary for a decision on issuance of construction permits. However, the river intake structure will be designed by the applicant to withstand the postulated standard project flood. Electric

motors and controls for intake pumps and traveling screens will be set above the standard project flood elevation. The plant can be shut down for an extended period of time by using either the mechanical draft cooling towers or the nuclear service water pond. Neither system relies on the river intake as a single source of water (see Section 9.2.5 for our evaluation of the bases for the design of the ultimate heat sink).

(3) The ground floor level of all plant substructures will be at least one foot above the high points of yard grade. In the event of blockage of site drains, the surface water will be conveyed away from plant buildings by natural flow processes with no potential for flooding the plant's safety-related facilities. The roofs of safety-related buildings will be provided with overflow scuppers to allow for runoff in the event of normal roof-drain blockage and for rainfall in excess of design values and up to the probable maximum value. In the event of scupper blockage, water will accumulate until it spills over the side walls. Roof penetrations on safety-related buildings will be waterproofed to above the maximum possible water level. We conclude that these considerations for site and roof drainage are acceptable.

(4) The nuclear service water pond will be formed by a seismic Category I dam to be constructed across McGowan Creek, a small tributary of the Broad River, immediately west of the plant area. The dissipation of heat from the nuclear service water will be normally accomplished by closed-cycle mechanical draft cooling towers. The pond will function as the ultimate heat sink if the towers are inoperable.

The pond will have a drainage area of approximately 1550 acres. The water level of the nuclear service water pond will be controlled by an ungated overflow spillway with an ogee crest at elevation 570.0 feet above mean sea level. The probable maximum flood resulting from the probable maximum precipitation on the pond drainage area, routed through the reservoir and emergency spillway, will produce a peak pool elevation of 582.8 feet above mean sea level plus a 0.1-foot wind setup and one-foot wave runup for a total dynamic water level of 583.9 feet above mean sea level, which is 6.1 feet below the top of dam elevation of 590.0 feet mean sea level. The ogee weir will be founded in rock and will have a transition through a 200-foot concrete-walled rock chute to a 1400-foot concrete-lined channel to the Broad River. Slope protection will be provided for the upstream and downstream slopes of the nuclear service water pond dam and also at the spillway outlet works. Dumped stone riprap will be placed from seven feet below maximum drawdown to the crest on the upstream slope. Downstream slopes will have riprap protection to the Broad River 100-year flood elevation of 525.0 feet above mean sea level. We have reviewed the analyses and proposed flood protection for the nuclear service water pond dam, and have concluded that the applicant's design bases are acceptable. However, we will require additional details of the design of the spillway and its appurtenant structures in order to verify that the probable maximum flood can be safely discharged. We will require that these design details be reviewed and approved by us prior to construction. The applicant has committed to this requirement.

(5) The alternate nuclear service water makeup pond will be formed by a separate non-seismic Category I dam on a small southeastern arm of the nuclear service water pond. The pond will provide an alternate source of makeup water for the nuclear service water cooling towers, which normally will be supplied with makeup from the intake sedimentation basin. The standard project flood, routed through the reservoir, resulted in a calculated maximum pond elevation of 58[illegible].0 feet above mean sea level which is three feet below the top of the the dam. Normally this capability would be acceptable for such an alternate source of water. However, this dam is located upstream of the nuclear service water pond dam, and it is essential that failure of the alternate nuclear service water pond dam does not pose a threat to the safety of the downstream seismic Category I nuclear service water dam. The applicant has stated that the seismic Category I nuclear service water dam will be designed to withstand the instantaneous failure of the alternate nuclear service water dam. We have also analyzed the failure of the alternate nuclear service water dam with the reservoir at elevation 590 feet above mean sea level and with the nuclear service water reservoir at elevation 583.9, and have concluded that the instantaneous failure of the alternate nuclear service water dam will not induce an overtopping failure of the nuclear service water dam. In Section 2.4.8.2.6 of the PSAR the applicant has committed to provide the details of the alternate nuclear service water makeup pond for our review and approval prior to construction.

2.4.3 Low River Consideration

Extended drought conditions in the Broad River Basin could induce loss of river water intake capability. However, the nuclear service water pond will be designed to provide sufficient storage to assure safe shutdown of the plant.

2.4.4 Groundwater

The proposed site lies within a groundwater region which is part of the Piedmont Groundwater Province. Groundwater in the area is derived entirely from local precipitation. The water is contained in the pores of the residual soils and in joints and cracks of the rock. There is a north-south groundwater ridge at the plant area, and groundwater flow is to the north, east, and west. The groundwater gradient in the plant area is about six to seven feet per 100 feet. Permeability is controlled by the extent and distribution of fractures in the bedrock and by the size and distribution of pores in the overlying soil. The applicant has made laboratory and field permeability tests and has determined values ranging from zero to about 5000 feet per year. Measured depths from the existing ground surface to the groundwater table on the ridges range from about 40 to 80 feet. However, the proposed plant grade will be at about existing groundwater level. The groundwater table is generally at or near the surface in valleys and draws near the site.

712 052

The applicant has analyzed two postulated spills of liquid radwaste to the groundwater regimes. In the first, it was postulated that one set of liquid radwaste storage tanks (with the potential for the largest source of radionuclides), located to the northwest of the reactors, would simultaneously rupture with no detention in the concrete vault in which the tanks reside. The contaminants would move in the groundwater northerly along the groundwater gradient and would break out as a spring at the toe of the plant yard fill. The contaminants would then travel as surface water to the Broad River where they would be further diluted by the historical minimum daily river flow of 224 cubic feet per second. This would result in a dilution factor of 11,200 and a travel time of 1.3 years with respect to the nearest water supply located 22 miles downstream from the site.

For the second postulated spill, it was assumed that one set of liquid radwaste storage tanks would simultaneously rupture and leak directly to the wall drain system where they would be pumped at maximum system capacity to the auxiliary holding pond. The applicant assumed the contaminants would be confined in the holding pond where they would not be a safety problem.

We do not concur with either of the applicant's analyses. Under the first postulation, we conclude that since the radius of influence for the underdrain system extends past the concrete vault any spill would be intercepted by the groundwater depression for the permanent dewatering system. It would then be pumped to the auxiliary holding pond which we assumed failed from either a seismic or flood event. This would result in a dilution factor of 6000 and travel time of 15 hours at the nearest downstream user 22 miles downstream. It is our position that the liquid radwaste-spill from the second postulation would be diluted initially by the auxiliary holding pond volume and would then seep through and under the dam and be diluted with the minimum average annual Broad River flow of record. We made independent calculations of the travel times and dilution factors based on our stated position and calculated a dilution factor of 36,000 and travel times of 4.7 years for strontium-90 and 8.8 years for cesium-134 and cesium-137 based on conservative ion exchange characteristics for the soil.

Considering dilution and radioactive decay over the above transit time, a rupture of the recycle holdup tank will give a known radionuclide concentration of less than 5×10^{-7} microcuries per milliliter at the nearest potable water source in the Broad River. This value is a fraction of the limits of 10 CFR Part 20, Appendix B, Table II, Column 2 for unrestricted areas.

2.4.5 Permanent Dewatering System

The applicant proposes to permanently lower the groundwater levels in the vicinity of safety-related structures by using a system of seismic Category I underdrains and exterior wall drains. The underdrains will consist of a series of interconnected flow channels spaced on 20-foot centers located under the foundation slabs. The exterior wall drains will consist of zoned filter materials around the walls, which will drain to a horizontal perforated pipe located at mat level. Both the underdrains and the perforated pipe will discharge to a sump located inside the auxiliary building from which the water will be pumped to the plant storm drains system for gravity flow to an auxiliary holding pond onsite. The underdrain system of connected flow channels will be located at the top of rock or at the top of the first level of fill concrete below the foundation slabs. Each channel will run the full length of the building excavation but will be closed at each end so that no sediment can be transported into it from backfill outside the walls. All channels in the grid system will drain by gravity through eight pipes to a 15-foot square sump located inside the auxiliary building. The exterior wall drains will be located around the exterior walls of the auxiliary and reactor building and will drain to the same sump as the underdrain system. No connection between the wall drains and the underdrain system will exist such that each drains to the sump through independent and separate conduits. The exterior wall drain system will consist of a zoned filter system which extends from five feet below yard grade to the bottom of the excavation. The continuous perforated pipe will extend around the perimeter of the building exterior walls at the bottom of the zoned wall filter. Two 120-gallon per minute seismic Category I pumps will maintain the water level automatically in the sump with each pump capable of handling the total computed flow of up to 35 gallons per minute per unit.

The applicant will include provisions in the design for monitoring of pump operation and visual inspection of drain outlets in the sump will provide assurance that the zoned filter, drains and pumps are functioning properly. Seismic Category I manholes, located along the exterior walls of the reactor and auxiliary buildings, will provide access to the perforated pipe in the zoned wall filter for inspection and cleanout. These manholes can be used for temporary installation of pumps in the unlikely event that groundwater rises in the wall drains. An inspection and monitoring procedure will be developed for both the construction and operation phase of the plant. Several observation wells will be located at strategic locations to monitor groundwater levels in the vicinity of the shield and auxiliary buildings and will be used to verify that the groundwater drawdown is effected as predicted and to establish its extent of influence in the yard area. These wells will be monitored periodically during construction for a sufficient period to verify that a steady state condition has been achieved. The details of the operational monitoring program will be provided during the operating license stage of our review.

The applicant states that design parameters used to size the dewatering system and to establish the monitoring program will be verified during construction excavation. The applicant has agreed that the currently proposed system would be modified or other groundwater drainage designs would be adopted in the event that the current design parameters are found to be substantially changed, as determined during construction excavation. For example, if the site soils or rocks are found to be more permeable, causing an increase in the design discharge, modifications such as increased pump size, or other designs would be implemented. The applicant has also agreed that the final design will be based on data gathered during the construction excavation, if the current design bases are inadequate.

We have reviewed the applicant's plans for providing monitoring programs during construction and operation and his commitment to appropriately modify the design if measurements show significantly higher groundwater flows than assumed for the preliminary design. We find that the monitoring program will provide sufficient data for design input, and conclude that an acceptable design can be provided for the measured groundwater flow.

In addition to the capability of the permanent dewatering system to handle normal groundwater flow, we asked the applicant to consider the effects of accidents and natural phenomena on the capability of the permanent dewatering system and the acceptability of the proposed structural design of the facility. The applicant had already considered the effects of infiltration of rainfall within the radius of influence assuming blockage of discharge pipes from the wall drains to the sump in the auxiliary building, but had not considered the effects on the sump in the absence of such blockage.

In considering accidents that could release fluids within the radius of influence, the applicant concentrated his assessment on a large source of water, the condenser circulating water system, and on sources that could be accidently released directly into the wall drain.

The applicant states that the failure of a circulating water system pipe inside the turbine building would cause water to be ponded to a depth of 13 feet above the turbine building floor. The wall of the adjacent auxiliary building facing the turbine building will be constructed as a seismic Category I wall up to a level of 13 feet-six inches above the turbine floor to prevent flow of the ponded water in the turbine building into the auxiliary building. In addition, the applicant proposes to place a grout curtain under this wall to reduce seepage to the underdrain system and to extend seismic Category I retaining walls outward from the auxiliary building to retain a column of low permeability soil as a barrier to flow of water from the turbine building around to wall drains along the sides of the auxiliary building.

The primary grout holes for the grout curtain below the auxiliary building substructure mat and the retaining wall will be spaced at 20-foot intervals. Secondary holes will split-space the primary grout holes. After the grout curtain is completed, with a maximum hole spacing of 10 feet center-to-center, four core holes will be drilled to verify the adequacy of the grout curtain. Along with visual inspection of the rock cores, the holes will be water tested to assure that the permeability of the grout curtain is less than the average permeability of continuous rock. The grout holes are to be split-spaced until the equality in permeability is attained. After completion of the grouting and testing, the four test holes will be cased and maintained for observation and testing throughout the life of the plant.

We conclude that the criteria for the design of retaining wall and placement of the grout curtain are acceptable and should result in an acceptable means of preventing leakage from the turbine building to the permanent dewatering system. In the event of a circulating water system pipe rupture outside of the turbine building, the applicant has stated that the results of an analysis predict that any additional water which will enter the dewatering system will be minimal, and normal groundwater levels will not be affected.

The applicant initially proposed as a design basis for subsurface hydrostatic loads, groundwater levels at the elevation of the underdrain system. During our review the applicant investigated the consequences of failures of some of the fluid containing tanks and piping within the radius of influence of the permanent dewatering system. Consequences of some of those failures which could release fluids directly into or near the permanent dewatering system were analyzed by the applicant.

The nuclear service water pipes will pass through the wall drain adjacent to the shield building. As described in Section 9.5.8 of this report, a moderate energy pipe crack within the wall drain would cause overflow of the sump and flooding of the auxiliary building floor, and in addition would cause a localized elevation of water in the wall drain by about 2.5 feet. The applicant in Section 2.4.13 of the PSAR has described the consequences of other accidents and additional design changes that were made to mitigate the consequences of the accidents. Although the accidents do not include all conceivable events that could result in excess flow into the sump, the applicant proposes to use the break of the nuclear service water pipe as the design basis event for evaluating sump overflow. It would appear that alternate designs, such as higher sump walls, could be readily implemented as a backup design feature if other sources result in unacceptable sump overflow. We conclude that the applicant's criteria for limiting sump overflow, or utilization of modifications to the preliminary design, if necessary, provide assurance that a design can be developed that will provide adequate flood protection for systems and components located in the shield and auxiliary buildings.

In response to our concerns about potential blockage of flow paths from the wall drain to the sump, the applicant has committed, as described in Section 3.8.5 of this report, to design external structural walls surrounded by wall drains and foundation floors to withstand as an extreme environmental load the hydrostatic load caused by postulated rebound of water in the wall drains to plant grade even though no specific mechanism for effecting such a rebound has been postulated. We conclude that this commitment is a conservative approach with respect to maximum design water level in the wall drain and is acceptable.

2.4.6 Ultimate Heat Sink

Independent sources of nuclear service water will be available to provide an adequate supply of cooling water to dissipate heat rejected during a postulated loss-of-coolant accident in one unit and a normal shutdown in the other two units. Each source will be separated so that failure of one does not cause failure of the other. Dissipation of waste heat in the nuclear service water will be normally accomplished by closed-cycle mechanical draft cooling towers. Two separate and redundant towers will comprise this

712 056

cooling system. Normal makeup for these towers will be furnished from the intake sedimentation basin. An alternate source of makeup water is provided by an alternate service water pond. Sufficient volume of water will be stored to assure availability of makeup for 30 days. A nuclear service water pond formed by a seismic Category I dam will provide sufficient storage of water for direct circulation to the nuclear service system with recirculation to the pond for heat rejection from the pond surface. On the basis of our review and independent analyses, we conclude that the hydrologic bases and design considerations for the ultimate heat sink are acceptable. Our overall conclusions on the ultimate heat sink are presented in Section 9.2.5 of this report.

2.5 Geology and Seismology

We have completed our review of the geology and seismology data contained in the PSAR. The seismology and geology review of the site addressed the geologic history of the region including physiographic, lithologic, stratigraphic and tectonic settings, as well as the subregional and site-specific geology and seismology. Staff geologists and seismologists visited the site. During these visits we examined the regional geology, bedrock and diabase dike exposures, and core borings and soil samples from the areas of the major structures and dam foundation areas.

Since the regional aspects which also apply to this site have been addressed extensively in other reviews and safety evaluations, including those for the William B. McGuire Nuclear Station and Catawba Nuclear Station, the main effort expended in our review dealt with resolving specific issues which might be of significance in relation to the proposed site.

We have concluded that the investigations performed by the applicant have been sufficient to adequately assess site geologic conditions. We reviewed available data and conducted discussions with geologic authorities familiar with the site area. These data indicate that there are no known geologic or seismic problems at the location of the site which would preclude the construction of the proposed nuclear power plants; however, due to the existence of ancient small-scale shears discovered in borings and test pits in the area, the geologic investigation program must be continued during excavation. The applicant has committed to this program (Appendix F to this report). The following paragraphs contain a summary of the geology, seismology and foundation engineering aspects of the proposed site.

2.5.1 Geology

The site is located in the Piedmont Physiographic Prov... of South Carolina about eight miles southeast of Gaffney, South Carolina, on the west side of the Broad River. The Piedmont Province, which trends northeast - southwest, extends from Alabama to New Jersey and is underlain by a complex sequence of deformed paleozoic metamorphic rocks, igneous rocks of Paleozoic age and sedimentary rocks of Triassic age. This province is bounded on the west by the Blue Ridge Province and on the east by the

Coastal Plain Province. A major structural geologic feature, the Brevard Fault Zone, is located between the Piedmont and Blue Ridge Provinces. Structurally, the Piedmont is characterized by large scale folds and ancient fault zones, or structural belts that trend northeast - southwest. These belts include from east to west: the Carolina State Belt, the Charlotte Belt, the Kings Mountain Belt, the Inner Piedmont Belt, and the Brevard Zone. The site is in the southern portion of the complex Kings Mountain Belt. Surface deposits are predominantly saprolitic soils and saprolite with scattered outcrops of metamorphosed bedrock. The surface material is underlain by mafic and felsic gneiss, schist, metaconglomerate, and quartzite. Potassium-argon dating of these rocks indicates that the last major episode of metamorphism occurred between 234 to 362 million years ago. A similar, but earlier event also occurred during the early to middle Paleozoic time span or about 400 million years ago. Because of the intense deformation which preceded or accompanied these regional metamorphic events, tight folds and minor shear zones were produced in the rocks of the region and their original sedimentary and volcanic fabric was altered, thus clouding their genesis and history. They are considered to be Precambrian and early Paleozoic aged sediments and volcanics deposited in a eugeosynclinal environment. The obscuration of geologic history, mentioned above, makes geologic mapping and determination of local and regional structural relationships difficult. This difficulty results from the region's low relief and from the thick cover of surface deposits that overlies bedrock in the area. Also, the rocks of this part of the Piedmont are highly jointed. Based on potassium-argon dating and field observation, the minor shear zones at the site are older than 170 million years and have displacements of no more than several inches. Several diabase dikes marking the last major tectonic event in the area have been injected into the rocks near the Cherokee site. These features have been sampled and dated; the ages range from 190 and 254 million years. Based on the detailed geologic, radiometric, and surface investigations, it can be said that there has been no tectonic activity at or near the site since the Jurassic about 150 million years ago.

Several features in the vicinity of the site have been described in the literature as major faults; however, examination of these features has shown no basis in fact for such a conclusion, or that there are alternative interpretations of the data which are more correct. Displacements of several feet resulting from minor faulting have been observed in a spodumene mine 13 miles north-northeast of the site and in a vermiculite mine 35 miles to the southwest. Regional geologic considerations and radiometric dating techniques indicate Triassic or Jurassic age assignments for the formation of these structures. Major tectonic structures in the region of the site are (1) the Gold Hill-Silver Hill Fault Complex, 40 miles east, dated by a pre-Triassic diabase at 238-254 million years; (2) the Jonesboro Fault, 60 miles east-southeast, which is associated with a diabase of Triassic-Jurassic age; (3) the problematic Brevard Zone about 50 miles west of the site whose development is believed to have ceased about Permian-Triassic time (225 million years ago); and (4) the Kings Mountain Compound Fold to the north of the site which was formed at the time or before the oldest shear zones and breccias were developed at the site.

Based on our review of the results of the applicant's site investigations and on the results of our own studies, we conclude that there are no known faults or other geologic structures in the immediate vicinity of the site which could be expected to localize earthquakes in the plant area.

2.5.2 Seismology

The applicant's review of literature and investigations of the site geology has not identified any geologically recent faulting in the site area. In addition, his work has shown that all reported evidence of possible major faulting within a 200-mile radius of the site is related to other geological phenomena, e.g., folding. Where minor faults have been found they have been dated as being geologically old and non-capable within the meaning of 10 CFR Part 100. As a result of these observations, no major recent faulting can be found within 200 miles of the site which could generate a large earthquake. Small earthquakes, however, have been observed in the Piedmont. None of these have been associated with faulting, although investigations of the depth and thoroughness that are characteristic of nuclear power plant siting investigations have not been made for the entire region. Such shocks are assumed to occur on small zones of weakness which are scattered at random throughout the Piedmont Province. The largest such shock was of Modified Mercalli intensity VII. The applicant and we have also considered both the consequences of a recurrence of the Charleston earthquake of August 31, 1886, 175 miles from the site, and the consequences of ground motion at the site from an earthquake on presently undetected major faults at distances greater than 200 miles from the site. Based on these considerations, we conclude that the 0.15g acceleration proposed by the applicant for the safe shutdown earthquake and the 0.08g acceleration for the operating basis earthquake are adequate for the bedrock at the site.

Accelerations greater than the bedrock acceleration might occur for structures founded on soil or fill overlying bedrock. Rather than designing for these effects the applicant proposes to lower the major foundations to bedrock and to design the plant for the safe shutdown earthquake of 0.15g and the operating basis earthquake of 0.08g.

On the basis of our analysis and evaluation, we conclude that there is reasonable assurance that there are no seismic or geologic related problems that would render the site unsuitable for the construction and safe operation of the proposed nuclear facilities.

2.5.3 Foundation Engineering

(1) Site

The plant site is in an area of low rounded hills which are divided by small drainage features that empty into the Broad River. Existing ground elevations at the site ranges from 550 to 650 feet about mean sea level. Plant finished grade in the vicinity of the major structures will be at elevation 590.0 feet above mean sea level, which is about 100 feet above the river level. The facility structure will be founded on rock and residual soil derived from rock weathering.

(2) Design Basis Earthquakes

The design basis earthquakes, consisting of an operating basis earthquake and a safe shutdown earthquake, are described in Section 2.5.2. For the safe shutdown earthquake, a ground motion with a peak acceleration at a zero period of 0.15g will be applied at the foundation level to rock supported structures. The response spectrum for this input motion at the foundation level will be in accordance with the provisions of Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants." For structures founded on soil, saprolite, or weathered rock, the design ground motion will be applied at the level of continuous rock and propagated upward to the foundation level. The applicant's preliminary soil amplification studies, indicate that the peak acceleration of 0.15 g at continuous rock level will be amplified by a factor of about two at the ground surface.

(3) Subsurface Conditions

The site is located in the Piedmont Physiographic Province and is characterized by a subsurface profile of residual soils and saprolite derived from the weathering of predominantly metamorphosed and deformed gneisses, schists and quartzite rocks. The texture of the soil-saprolite profile ranges from a thin surface layer of brown to red clayey silt residuum to mixed zones and layers of sandy silts and silty sands. There are quartz veins in the soil and rock profile which are pervious and difficult to detect by borings. The pervious condition does not present any problem for the foundation design of major structures, and the nuclear service water dam will be designed to provide for this condition by utilization of a core trench and a drainage blanket as discussed below under items (6) and (7).

The sandy silts comprise approximately 70 percent of the soils in the foundation and are predominantly derived from the weathering of gneiss and schist rock. They are mostly reddish tan in color, contain 10 to 60 percent mica and have 20 to 50 percent fines passing the number 200 sieve. The variations in the soil textures are gradual and are due to the changes in the composition of the parent rocks. These saprolitic soils gradually grade downward into felsic and mafic gneiss bedrock at a maximum depth of about 70 feet below the existing ground level. Bedrock is closely jointed.

Rock jointing coupled with variation in the mineralogical and chemical composition of the rock have caused large differential weathering effects at the site making the geologic profile very complex.

For engineering purposes, the applicant has chosen to define and classify materials as either soils, weathered rock, or continuous rock. Materials having a Standard Penetration Test resistance of less than 100 blows per foot are classified as soils. Soil materials extend to a maximum depth of 65 feet and include those materials which may be excavated using conventional earth moving equipment.

Partially weathered rock is defined as that rock below a horizon of dense saprolite having a Standard Penetration Test resistance greater than 100 blows per foot, and above continuous rock. The partially weathered rock zone ranges from a few feet to about 30 feet in thickness and will require ripping and light blasting during excavation. Continuous rock is hard rock having a Rock Quality Designation of at least 65 percent, a shear wave velocity greater than 5000 feet per second, and a minimum unconfined compressive strength of 1500 pounds per square inch.

The existing groundwater level in the plant area ranges between elevation 570 and 620 feet above mean sea level. Normal groundwater level during plant operation will be near the plant grade of 590 feet above mean sea level. However, the applicant has proposed to provide a permanent dewatering system to lower groundwater from this normal level to near foundation levels (Sections 2.4.4 and 2.4.5 of this report).

(4) Foundations for Structures in Main Plant Area

The shield and auxiliary buildings for each of the three units will have mat foundations founded on continuous rock below the zone of major weathering or on a thin layer of fill concrete over continuous rocks. The excavations for these structures will extend to a maximum depth of 50 feet below the level of continuous rock. In a few areas, such as that for the auxiliary building for Unit 2, the level of continuous rock will be slightly below foundation grade. In those locations, and other local areas where weathered rock is present below founding levels, the weathered rock will be excavated and replaced with fill concrete.

Each shield building for the three units will be supported on a 200-foot diameter circular mat. The auxiliary building for each unit, which occupies an area approximately 300 feet by 400 feet around the reactor building, will be supported on multiple mat foundations. Each building will be independently supported. Foundation loads for the mat foundations for the main plant structures are equal to or less than 11,000 pounds per square foot. The proposed rock foundation will adequately support the imposed loads and the applicant estimates that settlement will be negligible. However, we will require settlement monitoring using three monuments per structure and submittal of a record of the settlement history of these structures in the FSAR. Settlement monitoring should begin after the excavation is open and continue through and following construction.

Also, because of the complex geology and weathering profile of the rock at the site, we will require a test excavation in hard rock, during construction, to demonstrate that specifications and controls for blasting are adequate to assure that the reactor and auxiliary building excavations can be completed without unnecessary and unacceptable damage to the foundations. We will require that the test excavation area be of sufficient size to be representative of expected variation in geologic conditions, including rock type and the orientation of jointing and foliation planes. We will require that the applicant advise us when tests are complete and when the test area will be available for inspection, and that blasting records, including results and evaluation of tests, be submitted for our review. If the initial blasting methods used in the test area yield satisfactory foundation conditions, rock excavation can proceed for all safety-related structures using the same methods. If unsatisfactory results are obtained, the field data will be evaluated by the applicant and by us to develop suitable controls and criteria for blasting activity in critical foundation areas.

There is insufficient information in Section 5 of Appendix D to the PSAR, "Foundation Support," to complete our review. When sufficient information is available, we will provide our evaluation of foundations for above groundwater storage tanks and for buried diesel generator fuel oil tanks in a supplement to this report.

For design purposes, the at-rest lateral earth pressure coefficient of 0.5 will be increased by 32 percent to account for seismic effects. Lateral earth pressures due to compaction of backfill and construction of the dewatering system will be considered in the design.

(5) Permanent Dewatering System

As described in Section 2.4.5 of this report, the permanent dewatering system will consist of (a) a peripheral line drain (perforated pipe) which extends around the outside toe of the foundation of the auxiliary building complex, (b) a vertical blanket drain placed against the exterior surface of structural walls and, (c) a grid of drainage channels, spaced about 20 feet apart under the foundation mats. The drainage channels will be approximately four inches by six inches framed with wood, and placed on rock or on top of a leveling course of fill concrete. Where the channels are placed on fill concrete, drain holes penetrating rock joints, will be bored on a maximum spacing of eight foot centers. Also, a blanket of porous concrete will be placed at the rock contact beneath the fill concrete to provide a uniform water collection capability.

The applicant has committed to design seismic Category I structures for full hydrostatic loading conditions that would be attained in the absence of the dewatering system. The hydrostatic load will be considered as an extreme environmental or abnormal load (Sections 2.4.5 and 3.8.5 of this report).

Also, an underdrain performance monitoring system including the use of manholes, piezometers, observation wells, and alarm systems will be provided (Section 2.4.5).

We conclude that the design of the foundation engineering aspects of the permanent dewatering system are acceptable for the foundation conditions at the Cherokee Nuclear Station. Our conclusions on other aspects of the permanent dewatering system are reported in Sections 2.4.5, 3.8.5 and 9.5.8 of this report.

(6) Foundation Conditions for the Nuclear Service Water Facilities and Piping

The nuclear service water reservoir will be located west of the main plant area and will be formed by impounding water in a natural valley that extends from an elevation of about 510 feet above mean sea level in the flood plain to an average elevation of 600 feet above mean sea level on the edge of the reservoir basin. The main reservoir features will consist of an embankment dam, an uncontrolled chute type spillway, and low level intake and pump structures for conveying water from the reservoir to the power plant. The normal water level in the reservoir will be at elevation 570 feet above mean sea level and the crest of the dam will be at an elevation of 590 feet above mean sea level.

(a) Nuclear Service Water Dam

The maximum height of the embankment dam will be about 100 feet above the flood plain elevation. The maximum section will be founded on partially weathered rock, and on soils at both abutments. The final excavation limits for the dam foundation will be determined on the basis of Standard Penetration Test resistance, dynamic cone penetrometer resistance calibrated to the Standard Penetration Test, proofrolling, and field inspection by an experienced geotechnical engineer or geologist. Also, additional laboratory testing will be conducted on soil samples obtained after the excavations are open for verification of design strength assumptions. For the portion of the embankment supported on soil, the applicant has committed that foundation materials having a Standard Penetration Test resistance of less than 15 to 20 blows per foot will be considered unsuitable for support of the dam and will be removed and replaced with compacted embankment fill. Also, materials having shear strengths less than those assumed for design will be removed and replaced with compacted fill with adequate shear strength. On the abutments, where the embankment is to be founded on soil-saprolite, a core trench will be excavated to groutable rock. The trench will permit further exploration and inspection of subsurface conditions in order to determine the level from which foundation grouting should commence, and will subsequently be used for incorporating an internal drainage system in the embankment foundation.

(b) Nuclear Service Water Spillway

The uncontrolled chute spillway will have a 10-foot deep concrete ogee weir founded on hard rock with a total loading of 4,000 pounds per square foot.

The chute will slope on a two-percent grade and will discharge into the Ninety-Nine Islands at an elevation of 510.0 feet above mean sea level. It will be cut into dense saprolite and partially weathered rock. The section at the base of the spillway channel will be concrete lined, and a drainage blanket will be placed behind the walls for the collection of seepage.

712 063

Collected water will be discharged into the channel by drain holes through the concrete liner.

(c) Nuclear Service Water Piping and Intake, Discharge and Pump Structures

The nuclear service water intake and discharge structures will be founded on saprolite or on compacted engineered fill (Group I). These structures will have a total foundation loading of 1,000 pounds per square foot. The nuclear service water pump structures will be founded on partially weathered rock at an elevation of 550 feet above mean sea level with a total foundation loading of 2,500 pounds per square foot.

The nuclear service water intake and discharge pipes will be supported on saprolite, partially weathered rock, or compacted engineered fill. The applicant has committed to a test fill program during construction to develop specifications for the control of Group I engineered fill. We require that the final specifications for these fill materials, including controls for its placement, moisture, and compaction be provided to us for review and approval when they become available, but no less than 60 days prior to fill placement. We require that abrupt changes in the pipe support conditions which may cause differential settlement and pipe stress concentrations be avoided.

(d) Settlement Monitoring and Reporting

The settlement of the nuclear service water dam appurtenances, including the intake, discharge, and pump structures, and nuclear service water piping may be influenced by the reservoir loading, and changes in groundwater levels. Therefore, we will require settlement monitoring of these structures and careful evaluation of data through construction and for the life of the plant. We will require that sufficient instrumentation be provided for each structure to monitor total settlement, differential settlement, and tilt, and that settlement reference benchmarks be established on hard rock.

We will require that nuclear service water pipe connections not be made until it is determined by field measurement that settlements are within expected ranges, and that connections be sufficiently flexible to accommodate at least twice any additional settlement expected after the connections are made. We will require that the settlement history of these structures, together with details of how and when piping connections are made, be presented in the FSAR, and at the operating license stage of review will require that a program for monitoring settlement after construction be provided in the technical specifications.

712 064

(7) Nuclear Service Water Embankment Dam

The nuclear service water dam will be an earth embankment with a maximum height of 100 feet above its foundation level. The normal operating pool level will be 20 feet below the embankment crest. The crest width will be 40 feet and the upstream slope of the embankment will be one vertical on 3.5 horizontal while the downstream slope will be one vertical on three horizontal.

The embankment material will be mostly the micaceous sandy silt described under item (3) above. Borrow for the embankment will come from the excavations in the main plant area.

A zoned blanket drain will be placed on the prepared foundation beneath the downstream face of the embankment for the collection and control of seepage. The blanket drain will be extended down to groutable rock on the downstream face of the core trench excavation and extend up both abutments to the normal operating pool level of elevation 570 feet above mean sea level. To control through seepage, a chimney drain will be incorporated in the embankment at the upstream limit of the blanket drain. We will require that the minimum width of this drain be six feet to assure acceptable procedures and that it be constructed by keeping the level of the drain fill slightly above the embankment fill level to minimize contamination of the drain material during placement.

We will require that the near surface layer of silty clay - clayey silt residuum available from required excavations and borrow sources be used to construct an impervious embankment zone upstream and adjacent to the chimney drain and to the blanket drain in the core trench. We will require that this zone be at least 12 feet wide and extend from the base of the core trench, or top of grout curtain, to the crest of the embankment.

The applicant has committed to provide detailed excavation drawings and construction specifications relating to the design and construction of the nuclear service water embankment dam and its foundation for review prior to construction.

We will require that the applicant prepare the foundation excavation for an observation and provide two weeks notice for us. We will make at least one inspection of the nuclear service water dam foundation excavation during construction.

We conclude that the proposed design of the nuclear service water with our additional requirements, cited above, will be adequate for the geologic conditions of this site, as we know them. However, because the site geology is complex, we will require that foundation excavations be carefully inspected to determine if any local conditions exist which may adversely affect the performance of these structures, and if any such conditions are found, an assessment of the need for additional exploration and redesign.

3.0 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Conformance with General Design Criteria

The applicant has stated that Units 1, 2 and 3 of the proposed facility will be designed, constructed and operated in accordance with the Commission's General Design Criteria for Nuclear Power Plants (Appendix A to 10 CFR Part 50). On the basis of our review of the documentation supporting this commitment, we conclude that the proposed facility can be designed, constructed and operated to meet the requirements of the General Design Criteria. Discussions regarding compliance with each criterion are presented in Section 3.1 of the CESSAR and Section 3.1 of the PSAR.

3.2 Classification of Structures, Systems and Components

3.2.1 Seismic Classification

Our evaluation of the seismic classification of structures, systems and components important to safety which are within the scope of the nuclear steam supply standard reference system design is presented in Section 3.2.1 of Appendix A to this report. Therefore, the discussion below is limited to structures, systems and components which are within the scope of the balance of plant.

Safety-related structures, systems and components, which are within the scope of the balance of plant and are required to be designed to withstand the effects of a safe shutdown earthquake and remain functional, have been properly classified as seismic Category I items. 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 the guideline exposures of 10 CFR Part 100.

All other structures, systems and components that may be required for operation of the facilities are designed to other than seismic Category I requirements. Included in this classification are those portions of seismic Category I systems which will not be required to perform a safety function. Structures, systems and components important to safety that will be designed to withstand the effects of a safe shutdown earthquake and remain functional have been identified in an acceptable manner in Tables 3.2.1-1 through 3.2.2.4 of the PSAR. As noted in Section 3.2.1 of Appendix A to this report, acceptance of certain component cooling water lines to the reactor coolant pumps as Quality Group D designed to non-seismic Category I requirements is contingent upon favorable results of pump tests without component cooling water. These results will be evaluated during our review of the application for Final Design Approval.

712 066

The basis for our acceptance has been the conformance of the applicant's designs, design criteria and design bases for structures, systems, and components important to safety with the Commission's regulations as set forth in Criterion 2 of the General Design Criteria, and to Regulatory Guide 1.29, "Seismic Design Classification," staff positions, and industry standards.

We conclude that the safety-related structures, systems and components which are within the scope of the balance of plant and will be designed to withstand the effects of a safe shutdown earthquake and remain functional, have been properly classified as seismic Category I items in conformance with the Commission's regulations, the applicable Regulatory Guides, staff positions, and industry standards.

3.2.2 System Quality Group Classification

Our evaluation of the quality group classification of components important to safety which are within the scope of the nuclear steam supply standard reference system design is presented in Section 3.2.2 of Appendix A to this report. The discussion below is limited to structures, systems and components which are within the scope of the balance of plant.

Fluid system pressure retaining components important to safety, which are within the scope of the balance of plant, will be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed. The applicant has applied the American Nuclear Society classification system (Safety Classes 1, 2, 3 and non-nuclear safety) in accordance with American National Standards Institute Standard N18.2, to those fluid-containing components which 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 maintenance in a safe shutdown condition, and (3) to contain radioactive material. These classifications correspond to our Quality Groups A, B, C and D in Regulatory Guide 1.26, "Quality Group Classifications and Standards." These fluid systems have been classified in an acceptable manner in Tables 3.2.2-1, 3.2.2-2, 3.2.2-3 and 3.2.2-4 of the PSAR and on system piping and instrumentation diagrams in the PSAR.

The basis for acceptance in the staff's review has been conformance of the applicant's designs, design criteria, and design bases for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping and valves in fluid systems important to safety with the Commission's regulations as set forth in Criterion 1 of the General Design Criteria, the requirements of Codes specified in Section 50.55a of 10 CFR Part 50, and to Regulatory Guide 1.26, staff positions, and industry standards.

We conclude that the fluid system pressure-retaining components important to safety, which are within the balance of plant scope, that are designed, fabricated, erected and tested to quality standards in conformance with the Commission's regulations, the applicable Regulatory Guides, staff positions and industry standards, are acceptable.

3.3 Wind and Tornado Design Criteria

All seismic Category I structures exposed to wind forces will be designed to withstand the effects of forces imposed by the design wind. All seismic Category I systems and components located within these structures will be protected from the effects of the design wind. The design wind specified has a velocity of 95 miles per hour based on a recurrence interval of 100 years. The procedures that are used to transform the design wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gust factors are in accordance with the American Society of Civil Engineers paper No. 3269, "Wind Forces on Structures." This paper has been widely used and recognized and has been accepted for use in the design of recently-licensed nuclear power plants. The design wind loads will be combined with other applicable loads as discussed in Section 3.8 of this report.

All seismic Category I structures exposed to tornado forces and required to maintain their integrity for the safe shutdown of the facility, will be designed to withstand the effects of the design basis tornado. All seismic Category I systems and components located within these structures will therefore be protected from the effects of the design basis tornado. The design basis tornado conforms to the recommendations of Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," which specifies a tangential wind velocity of 290 miles per hour and a translational velocity of 70 miles per hour. The associated simultaneous atmospheric pressure drop is three pounds per square inch at a rate of two pounds per square inch per second.

The procedures that will be used to transform the tornado wind velocity into pressure loadings will be similar to those used for the design wind loadings, discussed above, except that no gust factors will be used and no change of velocity with height will be assumed. The pressure drop associated with the design tornado will be treated as a static load. The tornado missile effects will be determined using procedures discussed in Section 3.5 of this report. The total effect of the design basis tornado on seismic Category I structures will be determined by appropriate combinations of the individual effects of the tornado wind pressure, pressure drop and tornado-associated missiles. Tornado-generated loads will be combined with other applicable loads as discussed in Section 3.8 of this report.

All the plant structures not designed for the tornado effects will be investigated to assure that they will not fail to the extent that they might damage seismic Category I structures. The safety function and structural integrity of seismic Category I structures will thereby be assured.

We conclude that the procedures to be utilized to determine the loadings on seismic Category I structures induced by the design wind and by the design basis tornado specified for the facilities are acceptable, since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that, in the event of the occurrence of the design wind or the design basis tornado, the structural integrity of seismic Category I structures will not be impaired. Seismic Category I systems and components located within these structures will be adequately protected and will perform their intended safety functions. Conformance with these procedures is an acceptable basis for satisfying the requirements of Criterion 2 of the General Design Criteria.

3.4 Water Level (Flood) Design Criteria

The design basis flood levels resulting from the most unfavorable condition or combination of conditions that produce the maximum water level at the site are discussed in Section 2.4 of this report.

We have reviewed the hydrostatic and hydrodynamic effects associated with these flood levels and find them acceptable.

Although the plant grade is substantially above the river water level resulting from the probable maximum flood, flood conditions in the alternate service water pond could result in water levels in excess of the Cherokee plant grade which is at 590 feet above mean sea level. However, the elevated cooling tower yard and other areas higher than the flood levels will preclude flow from the pond to the plant yard.

The groundwater elevation in the vicinity of the reactor and auxiliary buildings, which contains safety-related equipment, will be lowered by a permanent dewatering system consisting of a wall drain system and an underdrain system. The designs of these systems and the bases for ground water levels to be specified for design are discussed in Sections 2.4 and 9.5.8 of this report. The resulting terms to be included in load combinations for these structures are discussed in Section 3.8.5 of this report.

We have reviewed the adequacy of the applicant's proposed design criteria and design bases to determine the loadings on seismic Category I structures induced by the highest design basis flood, or highest groundwater level, or other design basis events as described in Sections 2.4 and 9.5.8 of this report. We conclude that the design criteria and design bases for loadings due to water level on seismic Category I structures are acceptable.

3.5 Missile Protection

3.5.1 Missile Selection and Description

In Section 3.5 of the PSAR the applicant states that all seismic Category I structures and components, except those shielded from missiles will be designed for protection against missiles. The applicant states that he has considered (a) missiles identified in Table 3.5.3-1 of the CESSAR postulated to originate from equipment within the scope of the CESSAR, (b) tornado missiles, (c) turbine missiles, and (d) aircraft missiles.

We have concluded that nearby industrial, transportation, and military facilities pose no threat to safe plant operation (Section 2.2). We have categorized the other missiles considered by the applicant as (1) missiles generated by postulated failures of equipment within the facility, (2) missiles generated by postulated failures of the power conversion turbines and (3) missiles generated by postulated tornadoes. The results of our review of the applicant's missile selection is as follows:

(1) Facility Equipment Generated Missiles

Missiles that could be generated by postulated failures of equipment within the scope of the CESSAR are listed in Table 3.5.3-1 of the CESSAR. These include appurtenances to pressurized systems, e.g., nuts, bolts, studs, control rod drive assemblies and instrumentation nozzles. The possibility of missiles being generated due to overspeed of the reactor coolant pump is being reviewed by the staff as a generic issue. (Section 5.2.6 and Appendix C to this report.)

The results of our review of missiles selected within the scope of the CESSAR are reported in Section 3.5 of Appendix A to this report.

The applicant has not selected missiles from postulated failures of equipment within the scope of the PSAR and outside the scope of the CESSAR. However, he has committed to the CESSAR interface requirements stated in Section 3.5.4.1 of the CESSAR which includes requirements (1) to consider any potential missile within containment whose impact would lead to a loss-of-coolant accident or preclude systems within containment from carrying out their specified safety functions, (2) to consider any potential missile outside containment with a potential for preventing the system or equipment listed in Section 3.5.1 of the CESSAR from carrying out its specified safety functions, and (3) to consider any potential missile that could prevent conduct of safe shutdown, or prevent the plant from remaining in a safe shutdown condition. The applicant in Section 3.5 of the PSAR has committed to provide protection against internal or external missiles that could damage seismic Category I equipment and components and in Table 3.2.1-1 has committed to providing protection for the containment against equipment missiles.

We conclude that the applicant's criteria for missile protection is acceptable at this construction permit stage of review for selection of missiles due to failures of plant features outside the scope of the CESSAR.

(2) Turbine Missiles

The CESSAR in Section 3.5.1 includes a statement that the systems listed in Table 1.2-1 of the CESSAR are systems within the scope of the CESSAR whose damage by turbine missiles could have radiological consequences. The applicant in Section 3.5 of the PSAR states that protection is not provided specifically for turbine missiles on the basis that his analysis in Section 3.5.2.1 demonstrates an acceptably low probability of damage from turbine missiles. We did not evaluate the applicant's analysis but performed our own independent analysis.

Each of the three turbine generators will be arranged in a peninsular orientation relative to its respective containment. We reviewed the exposure of essential structures and systems of each unit to low trajectory turbine missiles postulated to occur in the other two units. We conclude that only about one-tenth of the turbine wheels could generate missiles that could strike safety-related structures and that the angles subtended by the exposed safety-related structures assure that for each postulated turbine missile the probability of a damaging strike is less than 10^{-3}. We consider that the probability of the occurrence of any destructive overspeed turbine missile is in the order of 4×10^{-5} per turbine-year and, hence, conclude that the overall probability of a damaging turbine missile strike is in the order of 10^{-7}. Since we consider this probability to be acceptably low, we conclude that the proposed locations and orientations of the turbine-generators will provide acceptable protection against potentially damaging low trajectory turbine missiles.

(3) Tornado Missiles

The applicant proposed seven postulated tornado missiles and described the bases for the proposed missile characteristics in Appendix 3A to the PSAR. We did not find that proposed missile spectrum acceptable. Subsequently in Amendment 28 to the PSAR the applicant in Table 3.5.3-1 proposed a revised tornado missile spectrum, which we find acceptable.

3.5.2 Structures, Systems and Components to be Missile Protected

The interface requirements stated in Section 3.5.4.1 of the CESSAR identify as systems to be protected: (1) inside containment the reactor coolant system and connecting systems, engineered safety feature systems, (2) outside containment the systems listed in Section 3.5.1 of the CESSAR and (3) all systems and equipment needed to conduct a safe plant shutdown, or to prevent the plant from remaining in a safe shutdown condition. As stated in Section 3.5 of Appendix A to this report, we have reviewed the interface requirements in Section 3.5.4.1 of the CESSAR and find them acceptable with respect to missile protection.

The applicant states in Section 3.5 of the PSAR that missile protection or redundancy will be provided for seismic Category I equipment and components such that internal or external missiles will not cause the release of significant amounts of radioactivity or prevent the safe and orderly shutdown of the reactor. In Table 3.2.1-1 of the PSAR the applicant has identified structures that will be protected against tornado missiles and structures that will be protected against equipment missiles.

712 071

We have concluded that the applicant's commitment to the CESSAR interface requirements and his additional commitments in Section 3.5 of the PSAR are acceptable at the construction permit stage of review for determining which systems will be protected against missiles.

3.5.3 Missile Barrier Design Procedures

The analysis of seismic Category I structures, shields and barriers to determine the effects of missile impact, will be accomplished in two steps. In the first step, for missiles generated by equipment failure, the potential damage that could be done by the missile in the immediate vicinity of impact will be determined. This will be accomplished by estimating the depth of penetration of the missile into the impacted structure. For concrete structures, the modified Petry equation will be used to determine the extent of missile penetration. For steel structures, formulas developed by the Stanford Research Institute for estimation of penetration of missiles will be used. These formulas are widely used and recognized and were used on recently licensed plants. Furthermore, secondary missiles will be prevented by fixing the target thickness well above that determined for penetration.

For tornado missiles, the applicant has committed by Amendment 28 to the PSAR to design walls and roofs exposed to tornado missiles to thicknesses shown in Table 3.5.3-2 of the PSAR. We conclude that this procedure is acceptable for tornado missiles and that local damage analyses of walls so sized need not be made for tornado missiles. However, analyses to predict overall structural response as described in the following paragraph is necessary. By Amendment 28 to the PSAR the applicant committed in Section 3.5.4 of the PSAR to demonstrate acceptable overall structural response for the tornado missile spectrum shown in Table 3.5.3-1 of the PSAR.

In the second step of the analysis, the overall structural response of the target when impacted by a missile will be determined using established and acceptable methods of impactive analysis. The load of missile impact, whether the missile is environmentally generated or accidentally generated within the plant, will be combined with other applicable loads as discussed in Section 3.8 of this report.

The use of these procedures provides reasonable assurance that, in the event of design basis missiles striking seismic Category I structures or other missile shields and barriers, the structural integrity of structures, shields and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected by these structures will, therefore, be adequately protected against the effects of missiles. Conformance with these procedures is an acceptable basis for satisfying the requirements of Criterion 4 of the General Design Criteria.

We conclude that the design procedures that will be utilized to determine the effects and loadings on seismic Category I structures, barriers and missile shields induced by design basis missiles selected for the plant are acceptable, since these procedures represent accepted practice for engineering design to assure that the structures or barriers are adequately resistant to the effects of missile impacts.

3.6 Criteria for Protection Against Dynamic and Environmental Effects Associated With the Postulated Rupture of Piping

3.6.1 Protection Against Dynamic and Environmental Effects Associated with the Postulated Rupture of Piping Inside Containment

Our safety evaluation of the criteria and methods for protection against the effects of postulated ruptures of the reactor coolant system loop piping which are within the scope of the nuclear steam supply system is presented in Section 3.6 of Appendix A to this report. In addition, the applicant has incorporated provisions in the design of the piping systems which are within the scope of balance of plant that are generally consistent with Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment." We conclude that exceptions to Regulatory Guide 1.46 proposed by the applicant that are delineated in Table 3.6.1-3 of the PSAR are acceptable.

These provisions for protection against the dynamic effects associated with pipe ruptures and the resulting discharging coolant provide acceptable assurance that, in the event of the occurrence of the combined loadings imposed by an earthquake of the magnitude specified for the safe shutdown earthquake and a concurrent single pipe break of the largest pipe at one of the design basis break locations, the following conditions and safety functions will be accommodated and assured:

(1) The magnitude of the design basis loss-of-coolant accident cannot be aggravated by potentially multiple failures of piping.

(2) The reactor emergency core cooling systems can be expected to perform their intended function.

(3) The containment structure's leak-tight integrity can be expected to be maintained in order to contain within the leakage limits of the containment, any radioactive materials released from the discharging coolant into the containment atmosphere.

On the basis of our review, we conclude that the criteria that will be used for the identification, design and analysis of piping systems, where postulated breaks may occur, constitute an acceptable design basis in meeting the applicable requirements of Criteria 1, 2, 4, 14 and 15 of the General Design Criteria.

3.6.2 Criteria for Protection Against Dynamic and Environmental Effects Associated with the Postulated Rupture of High Energy Piping Outside Containment

The proposed design will accommodate the effects of postulated pipe breaks and cracks in high energy fluid piping systems outside containment with respect to pipe whip, jet impingement and resulting reaction forces, and environmental conditions. The general arrangement and the layout of high energy systems will utilize the possible combinations of physical separation, pipe enclosures, pipe whip restraints and equipment shields.

The criteria to be followed in the design of the piping systems and associated components and structures will be in accordance with those contained in Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment" with exceptions identified in Table 3.6.1-3 of the applicant's PSAR. We have reviewed these exceptions and find them acceptable for the proposed facility.

The applicant will analyze high energy piping systems for the effects of pipe whip, jet impingement, and environment on safety-related systems and structures. For moderate energy systems, the jet and environmental effects due to critical cracks will also be considered.

The plant design basis will include the ability to sustain a postulated high energy pipe break accident coincident with a single active failure and retain the capability for safe cold shutdown. For postulated pipe failures, the resulting environmental effect will not preclude the habitability of the control room, the accessibility of other areas that have to be manned during an accident condition, and the loss of function of electric power supplies, controls and instrumentation needed to complete a safety action.

We conclude that the design criteria and bases to be used for protection of essential systems and components from a postulated failure of piping outside the containment are acceptable.

3.7 Seismic Design

Our evaluation of the seismic design of systems and components within the scope of the standard reference system design is presented in Section 3.7 of Appendix A to this report. Our discussion below is limited to structures, systems and components within the scope of the balance of plant.

3.7.1 Seismic Input

The seismic design response spectra to be applied in the design of seismic Category I structures, systems and components comply with the recommendations of Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants."

The specific percentage of critical damping values to be used in the seismic analysis of seismic Category I structures, systems and components are in conformance with Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

The synthetic time history to be used for the seismic design of seismic Category I plant structures, systems and components is adjusted in amplitude and frequency content to obtain response spectra that envelop the design response spectra specified for the site.

712 074

Conformance with the recommendations of Regulatory Guides 1.60 and 1.61 provides reasonable assurance that the systems and components will be adequately designed to withstand the consequent seismic loadings associated with the operating basis earthquake and safe shutdown earthquake accelerations.

We conclude that the applicant's proposed seismic input criteria are acceptable for seismic design.

3.7.2 Seismic System and Subsystem Analysis

The scope of our review for the seismic system and subsystem analysis for the balance of plant included the following: (1) the seismic analysis methods for all seismic Category I structures, systems and components, (2) procedures for modeling, (3) seismic soil-structure interaction, (4) the development of floor response spectra, (5) the inclusion of torsional effects, (6) seismic analysis of seismic Category I dams, (7) the evaluation of seismic Category I structure overturning, (8) design criteria and procedures for evaluation of interaction of non-seismic Category I structures and piping with seismic Category I structures and piping, and (9) the effects of parameter variations on floor response spectra.

The system and subsystem analysis will be performed by the applicant on an elastic basis. Modal response spectrum multi-degree-of-freedom and time history methods form the basis for the analysis of all major seismic Category I structures, systems and components. When the modal response spectrum method is used, governing response parameters will be combined by the square-root-of-the-sum-of-the-squares rule. However, the absolute sum of the modal responses is used for modes with closely spaced frequencies.

The square-root-of-the-sum-of-the-squares of the maximum codirectional responses will be used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. Floor spectra inputs to be used for design and test verifications of structures, systems, and components will be generated from the time history method, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis will be employed for all structures, systems, and components where analyses show significant structural amplification in the vertical direction. Torsional effects and stability against overturning will also be considered.

The finite element approach will be used for the analysis of seismic Category I dams. This approach will take into consideration the time history of forces and the behavior of the deformation of the dam due to the earthquake, and the applicable stress-strain relations will be used.

In Amendment 29 to the PSAR, the applicant will add a commitment on interaction of non-seismic Category I structures with seismic Category I structures in accordance with the provisions of Section 3.7.2, II 8 of "Standard Review Plan For the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-75/087.

The applicant will verify that the response spectra obtained at the interface of the structure and the nuclear steam supply system will be less than those specified in the CESSAR.

We conclude that the seismic system and subsystem analysis procedures and criteria proposed by the applicant provide an acceptable basis for the seismic design of seismic Category I structures, systems and components.

3.7.3 Seismic Instrumentation Program

The installation of the specified seismic instrumentation in the reactor containment structure and at other seismic Category I structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data to be obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. We have determined that the proposed seismic instrumentation program complies with the provisions of Regulatory Guide 1.12, "Instrumentation for Earthquakes," and is, therefore, acceptable.

3.8 Design of Seismic Category I Structures

3.8.1 Steel Containment

The containment will consist of a spherical, free-standing steel shell located within a separate, reinforced concrete shield building. The containment will be designed, fabricated, constructed and tested as a Class MC vessel in accordance with Subsection NE Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, 1971 Edition. Loads will include an appropriate combination of dead and live loads, thermal loads, seismic and loss-of-coolant accident induced loads, including pressure and jet forces. A seismic Category I concrete shield building will protect the steel containment from the effects of wind and tornadoes and various postulated accidents occurring outside the shield building.

The analysis of the containment will be based on the elastic thin shell theory. The allowable stress and strain limits for the various loading conditions are generally those delineated in the applicable sections of Subsection NE of the ASME Code, Section III supplemented by the Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components." After the completion of construction and prior to operation, the containment will be subjected to structural proof tests, including hydrostatic, pneumatic or leak tests in accordance with Subsection NE of the ASME Code, Section III and Regulatory Guide 1.57.

The criteria to be used in the analysis, design, construction and testing of the steel containment structure to account for the loading and conditions that are anticipated to be experienced by the structure during the service lifetime, are in conformance with the acceptable rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE, Class MC Components. By letter dated February 8, 1977, the applicant has committed to apply the explicit load combinations delineated in Section 3.8.2, paragraph II 3 of the "Standard Review Plar for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-75/087, November 24, 1974. The criteria used in the analysis, design, and construction of the steel containment structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are generally in conformance with established criteria, codes, standards, and guides which we find to be acceptable.

The use of these criteria as defined by applicable codes, standards, and guides; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques, and the testing and in-service surveillance requirements, provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within and outside the containment, the steel containment will withstand the specified conditions without impairment of structural integrity or safety function. Conformance with these criteria constitutes an acceptable basis for satisfying the applicable requirements of Criteria 2, 4, 16, and 50 of the General Design Criteria.

3.8.2 Concrete and Structural Steel Internal Structures

The containment interior structures will consist of a shield wall around the reactor, secondary shield walls and other interior walls, compartments and floors. The interior structures will be designed in accordance with the American Concrete Institute (ACI)-318-71 Code for concrete and the American Institute of Steel Construction (AISC) specifications 7th Edition for structural steel. The load factors, the maximum allowable stresses and the load combinations of both of these codes have been modified in accordance with positions that we have developed to adopt them to the conditions encountered in the design of nuclear plants.

The applicant has considered those loads which may act on the structure during its lifetime, such as dead and live loads, accident-induced loads, including pressure and jet loads, and seismic loads. The loa combinations to be used cover all postulated events and include all loads which may act simultaneously. In the design of concrete interior structures, the strength design method will be used.

The criteria to be used in the design, analysis, and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetime are in conformance with established criteria, and with codes, standards, and specifications which we find to be acceptable.

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occuring within the containment, the interior structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Conformance with these criteria constitutes an acceptable basis for satisfying the applicable requirements of Criteria 2 and 4 of the General Design Criteria.

3.8.3 Other Seismic Category I Structures

Seismic Category I structures other than the containment will include the containment shield building, the service water intake structures, the auxiliary building which includes the fuel handling area and the control room. With the exception of the containment shield building which is cylindrical in shape with a hemispherical dome the seismic Category I structures other than containment will be predominantly rectangular type structures consisting of slabs, beams, walls and columns.

The major code to be used in the design of concrete seismic Category I structures will be the ACI 318-71, "Building Code Requirements for Reinforced Concrete." For steel seismic Category I structures, the AISC specification, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," will be used. The load factors, the maximum allowable stresses and the load combinations of both of these codes have been modified in accordance with the positions that we have developed to adopt them to the conditions encountered in the design of nuclear plants.

The concrete and steel seismic Category I structures will be designed to resist various combinations of dead loads; live loads; environmental loads including winds, tornadoes, operating basis earthquake and safe shutdown earthquake; and loads generated by postulated ruptures of high energy pipes such as reaction and jet impingement forces, compartment pressures, and impact effects of whipping pipes.

The design and analysis procedures that will be used for these seismic Category I structures are the same as those approved on previously-licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-71 Code and in the AISC specification for concrete and steel structures, respectively.

The various seismic Category I structures will be designed and proportioned to remain within limits that we have established under the various load combinations.

The criteria to be used in the analysis, design and construction of the seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime, are in conformance with established criteria, codes, standards, and specifications which we find to be acceptable.

The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control and special construction techniques; and the testing and in-service surveillance requirements, provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of their structural integrity in the performance of their safety function. Conformance with these criteria codes, specifications, and standards constitutes an acceptable basis for satisfying the applicable requirements of Criteria 2 and 4 of the General Design Criteria.

3.8.4 Foundations

Foundations of seismic Category I structures are described in Section 3.8.5 of the PSAR. Primarily, these foundations will be reinforced concrete of the mat type. These foundations, in most cases, will be supported directly on sound rock or fill concrete that extends to sound rock. The major code to be used in the design of these concrete mat foundations is ACI 318-71. These concrete foundations will be designed to resist various combinations of dead loads, live loads, environmental loads including winds, tornadoes, operating basis earthquake and safe shutdown earthquake, and loads generated by postulated ruptures of high energy pipes.

The design and analysis procedures that will be used for these seismic Category I foundations are the same as those approved on previously-licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-71 Code. The various seismic Category I foundations will be designed and proportioned to remain within limits that we have established under the various load combinations. These limits are, in general, based on the ACI 318-71 Code modified as appropriate for load combinations that are considered extreme. The materials of construction, their fabrication, construction and installation, will be in accordance with the ACI 318-71 Code.

The criteria used in the analysis, design and construction of plant seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications which we find to be acceptable.

The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; and the materials, quality control and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated events, seismic Category I foundations will withstand the specified design conditions

712 079

without impairment of structural integrity and stability or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying in part the requirements of Criteria 2 and 4 of the General Design Criteria.

3.8.5 Loads Due to Failure of Permanent Dewatering System

As described in Section 2.4.5 of this report, exterior walls of structures located within the permanent dewatering system will be designed to withstand the full hydrostatic pressure resulting from full ground water level rebound to plant grade following a postulated blockage of the lines to the dewatering system sump. This load will be treated as an extreme environmental load and will be combined with other normal operating loads as shown in Table 3.8.1-2 of Amendment 27 to the PSAR.

The hydrostatic load resulting from postulated leakage cracks in the nuclear service water pipe will be defined as an abnormal load, P_a, and included in the several load combinations of Table 3.8.1-2 of the PSAR.

The exterior walls and foundation mats of the shield and auxiliary buildings will be designed for these load combinations. We conclude that this approach is acceptable for the design basis events proposed by the applicant.

3.9 Mechanical Systems and Components

3.9.1 Dynamic Systems Analysis and Testing

Our evaluation of the criteria, testing procedures and dynamic analysis employed to assure structural and functional integrity of piping systems, mechanical equipment, and reactor internals, which are within the scope of the standard reference system design, is presented in Section 3.9 of Appendix A to this report. Therefore, the discussion below is limited to piping systems and mechanical equipment which are within the scope of the balance of plant.

The applicant will perform a preoperational piping vibrational and dynamic effects test program to confirm that dynamic loadings on piping from operational transients conditions have been properly accounted for in the design and analysis of piping systems and restraints classified as ASME Code Class 1 and 2 components. This program will provide adequate assurance that the piping and piping restraints of the systems will be designed to withstand vibrational dynamic effects due to valve closures, pump trips and operating modes associated with the design operational transients. The tests, as planned, will develop loads similar to those experienced during reactor operation. A commitment to proceed with such a program constitutes an acceptable design basis at the construction permit stage of review in fulfillment of the applicable requirements of Criterion 15 of the General Design Criteria.

The applicant has proposed acceptable dynamic testing and analysis procedures to confirm the adequacy of all seismic Category I mechanical equipment, including their supports, to function during and after an earthquake of magnitude up to and including the safe shutdown earthquake at the site. Subjecting the equipment and supports to these dynamic testing and analysis procedures provides reasonable assurance that, in the event of an earthquake at the site, the seismic Category I mechanical equipment will continue to function during and after the seismic event. We conclude that implementation of these dynamic testing and analysis procedures constitutes an acceptable basis for satisfying the requirements of Criteria 2 and 14 of the General Design Criteria.

3.9.2 ASME Code Class 2 and 3 Components

All seismic Category I pressure-retaining systems, components and equipment outside of the reactor coolant pressure boundary, including active pumps and valves, are to be designed to sustain normal loads, anticipated transients, the operating basis earthquake, and the safe shutdown earthquake within stress limits which are comparable to those outlined in Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components".

The specified design basis combinations of loading, as applied to the design of the safety-related Code Class 2 and 3 pressure-retaining components in systems classified as seismic Category I, provide reasonable assurance that in the event: (1) an earthquake should occur at the site, or (2) an upset, emergency or faulted plant transient should occur during normal plant operation, the resulting combined stresses imposed on the system components may be expected not to exceed the allowable design stress and strain limits for the materials of construction.

Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without gross loss of structural integrity. The design load combinations and associated stress and deformation limits specified for all ASME Code Class 2 and 3 components, including the active pumps and valves, constitute an acceptable basis for design in satisfying Criteria 1, 2 and 4 of the General Design Criteria.

The applicant has agreed to utilize an operability assurance program, in addition to the limits on stress and deformation, to qualify active ASME Class 2 and 3 seismic Category I pumps and valves. Such a program will include component testing, or a combination of tests and predictive analysis supplemented by seismic qualification testing of motors, operators, and components appendages to provide assurance that such components can withstand postulated seismic loads in combination with other significant loads without loss of structural integrity, and can perform the "active" function (i.e., valve closure or opening or pump operation) when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated. We have concluded that this commitment to develop and utilize a component operability assurance program is acceptable and constitutes an acceptable basis at the construction permit stage of review for assuring the operability of ASME Code Class 2 and 3 active pumps and valves.

The criteria to be used in developing the design and mounting of ASME Class 2 and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed the allowable design stress and strain limits for the materials of constuction. Limiting the stresses under the loading combinations associated with the actuation of the pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function.

The criteria originally proposed to be used for the design and the installation of ASME Class 2 and 3 overpressure relief devices constitute an acceptable design basis in meeting the applicable requirements of Criteria 1, 2, 4, 14 and 15 of the General Design Criteria, and are consistent with the provisions of Regulatory Guide 1.67 "Installation of Overpressure Protection Devices." By letter of February 8, 1977, the applicant proposed an exception to Regulatory Guide 1.67 that consists of an exception to ASME Code Case 1569. We will not complete our review of this matter until after the ASME acts on this proposal. We will require that prior to a decision on issuance of construction permits, the applicant reconfirm that his design and installation of ASME Class 1 overpressure relief devices will be in accordance with the provisions of Regulatory Guide 1.67, including the Guide's adoption of ASME Code Case 1569.

3.10 Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment

Our evaluation of the design of seismic Category I instrumentation and electrical equipment within the scope of the standard nuclear steam supply reference system design is presented in Section 3.10 of Appendix A to this report. Therefore, the discussion below is limited to seismic Category I instrumentation and electrical equipment within the scope of the balance of plant.

Operability of the instrumentation and electrical equipment is essential to assure the capability of such equipment to initiate protective actions in the event of a safe shutdown earthquake, as necessary, for the operation of engineered safety features and standby power systems. The proposed seismic qualification program, which will be implemented for seismic Category I instrumentation and electrical equipment and supports within the balance of plant scope, will provide assurance that such equipment may be expected to function properly and that structural integrity of the supports will be maintained during the excitation and vibratory forces imposed by the safe shutdown earthquake under the conditions of post-accident operation.

For Class IE equipment within the balance of plant scope, the applicant has stated that the purchase specification requirements will comply with those provided in Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1971, "Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations." Additionally, these requirements will be supplemented by multi-frequency excitation and multi-axis testing in accordance with Standard Review Plan Section 3.10.

712 082

We conclude that the above commitments for the seismic qualification of Class IE equipment comply with staff technical positions and are acceptable. This program constitutes an acceptable basis for satisfying the requirements of Criterion 2 of the General Design Criteria.

3.11 Environmental Design of Mechanical and Electrical Equipment

The applicant has adequately identified the safety-related (Class IE) equipment and, for each item, the environmental design basis, the definition of normal and postulated accident environments, and the required time of post-accident operability.

The applicant proposes to qualify Class IE equipment in accordance with IEEE Standards 323-1974, 382-1974, 317-1972, and 334-1971 and has presented the same additional comments on IEEE Std 323-1974 as are found in the CESSAR. We have interpreted these comments as a commitment to meet the requirements of IEEE Std 323-1974 without exception.

We informed the applicant that we had reviewed the descriptive information presented by the application in support of the above commitment and noted that the temperatures for which the instrumentation and control equipment inside of containment will be qualified are given in Table 3.11.2 of the PSAR. The maximum temperatures are 407 degrees Fahrenheit for 60 seconds and 350 degrees Fahrenheit for 9 minutes. Our analysis of the worst case accident containment temperatures shows a peak at 403 degrees Fahrenheit and temperatures that exceed 350 degrees Fahrenheit for 72 seconds. We required, and the applicant has committed to conduct appropriate testing to ensure that the staff's calculated worst case environmental conditions are used as the basis for equipment qualification. The test results will be documented during the operating license stage of our review. We also required that the applicant provide an implementation program including the test methods and documentation requirements for meeting IEEE Std 323-1974, as required in Section 3.11 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," NUREG-75/094, within six months of any issuance of construction permits.

As reported in Section 7.6-1 of Appendix A to this report Combustion Engineering has stated that all class IE equipment in Combustion Engineering's scope will be qualified for use under specified environmental service conditions in accordance with IEEE 323-1974, without exception. For Class IE equipment which is not within the Combustion Engineering scope of supply the applicant states,

> "The qualification method and qualified life will be established at the time of purchase by determining whether specified equipment has or will have had previous operating experience, is already qualified with satisfactory qualified life to IEEE 323-1974, or will require on-going qualification. The method of qualification for each item of equipment can be determined as specific manufacturers are known, thereby, establishing known operating history, exact vendor qualification method in compliance with appropriate standards, or the necessity of on-going

qualification. The schedule for this engineering effort will generally coincide with equipment purchasing schedules. Approximately six months after issuance of construction permits for the proposed facility, the equipment listing and other available information will be compiled into a technical report and filed with the Commission for review."

Discussions between the staff and the applicant have established that the other information which will be in the technical report will include a statement as to how each piece of equipment has been or will be qualified and analytical methods and results or test results, if available, when the report is prepared. This commitment is acceptable to us.

With regard to qualification methods for equipment which cannot be pre-aged the applicant has stated,

"For equipment with a qualified life less than full plant life, on-going qualification program will utilize the operating history of similar equipment at other plants in conjunction with in-plant baseline parameter monitoring. This method of on-going qualification will be based on established periodic testing below:

(1) Establish baseline data for the equipment at factory checkout or during testing following installation.

(2) Select appropriate indicator parameters for on-going monitoring to be compared with baseline data.

(3) Establish initial on-going test and surveillance frequencies to maintain operability based on vendor information, operating history, and/or analysis.

(4) Determine degradation level of indicator parameters to be allowed before corrective action is taken. Corrective action will be maintenance, modification, or replacement."

Discussions between the staff and the applicant have established that the on-going qualification program will only be used in-plant for the qualification of equipment in locations where the normal operating environment (including electrical supply from a Class IE source) will be the same as the design bases accident environments. Because the applicant understands that all Class IE equipment must be tested at the conditions for which it is to be qualified and the indicator parameters must contain suitable margin, we find the use of an on-going qualification program acceptable.

712 084

As reported in Section 7.6.1 of Appendix A to this report, we conclude that the proposed criteria for the qualification of Class IE equipment in the CESSAR can facilitate development of a qualification program consistent with the objectives established in IEEE Std 323-1974 and that the commitment described provides an acceptable basis for the Preliminary Design Approval of the Class IE equipment qualification program.

085
712

4.0 REACTOR

Our evaluation of the reactor design is presented in Section 4.0 of Appendix A to this report.

5.0 REACTOR COOLANT SYSTEM

5.1 Introduction

Our evaluation of the reactor coolant system is presented in Section 5.0 of Appendix A to this report. Therefore, our discussions below are specifically related to the appropriate portions of the balance of plant. The section numbering system used in this section is based on the numbers in Section 5.0 of Appendix A to this report that deal with the same subject matter.

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary Components

Components of the reactor coolant pressure boundary, as defined by 10 CFR Part 50, Section 50.55a, have been properly identified and classified as American Society of Mechanical Engineers (ASME) Section III, Code Class I components in Table 5.2-1 of the CESSAR. These components within the reactor coolant pressure boundary will be constructed in accordance with the requirements of the applicable codes and addenda as specified by the rules of 10 CFR Part 50, Section 50.55a, Codes and Standards. We have concluded that construction of the components of the reactor coolant pressure boundary in conformance with the Commission's regulations provides reasonable assurance that the resulting quality standards will be commensurate with the importance of the safety function of the reactor coolant pressure boundary, and is acceptable.

The ASME Code Cases specified in Table 5.2-6 of the CESSAR, whose requirements will be applied in the construction of pressure-retaining ASME Section III, Class 1 components within the reactor coolant pressure boundary (Quality Group Classification A), are in accordance with those code cases stated in Regulatory Guides 1.84, "Code Case Acceptability--ASME Section III Design and Fabrication," and 1.85, "Code Case Acceptability--ASME Section III Materials," that are generally acceptable to us. We conclude that compliance with the requirements of these code cases, in conformance with the Commission's regulations, will result in a component quality level that is commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

5.2.2 Overpressurization Protection

The reactor coolant system design relies upon the combined action of the pressurizer safety valves, the steam system safety valves, and the reactor protection system for overpressurization protection. The standard reference system design scope includes the pressurizer safety valves and the reactor protection system. Our evaluation of these systems is contained in Section 5.2.2 of Appendix A to this report. The steam and feedwater system piping and valves, including the relief and safety valves to protect the steam generator shell side against overpressurization are within the balance of plant scope.

We have evaluated the design commitments with respect to the interface requirements and preliminary overpressurization protection analysis and assumptions used in the accident analyses. Based on our review, we conclude that the design is acceptable for the construction permit stage of review. However, equipment malfunction or operator error when the reactor coolant system is water-solid during startup or shutdown could result in inadvertent reactor vessel overpressurization. The staff will require that the applicant provide acceptable equipment design modification in the Final Safety Analysis Report for the proposed facility which will preclude such overpressure events. We will require a commitment to this requirement prior to a decision on issuance of construction permits.

The relief requirements for the shell side of the steam generator are specified in the CESSAR as design interface requirements that must be met by the user. The applicant in Section 1.9.3.3 of the PSAR shows no exception to these interface requirements.

In Section 5.2.2 of Appendix A to this report we conclude that the criteria used for the design and installation of ASME Class 1 overpressure relief devices are consistent with Regulatory Guide 1.67, "Installation of Overpressure Protection Devices," and constitute an acceptable basis in meeting the applicable requirements of Criteria 1, 2, 4, 14 and 15 of the Commission's General Design Criteria. By letter of February 8, 1977, the applicant proposed an exception to Regulatory Guide 1.67 that consists of an exception to ASME Code Case 1569. We will not complete our review of this matter until after the ASME acts on this proposal. We will require that prior to a decision on issuance of construction permits the applicant reconfirm that his design and installation of ASME Class 1 overpressure relief devices will be in accordance with the provisions of Regulatory Guide 1.67, including the Guide's adoption of ASME Code Case 1569.

5.2.6 Pump Flywheel

In Section 5.2.6 of Appendix A to this report, we conclude that the CESSAR committed conformance with the recommendations of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," constituted an acceptable basis for satisfying the requirements of Criterion 4 of the Commission's General Design Criteria. In Amendment 28, the applicant in Table 1.7-1 of the PSAR has committed to partial compliance with the inservice inspection recommendations of Regulatory Guide 1.14. We will report the results of our evaluation of these exceptions in a supplement to this report. We will require conformance with Regulatory Guide 1.14 or a demonstration that an equivalent level of safety is provided by the alternative proposal prior to a decision on issuance of construction permits.

Criterion 4 of the General Design Criter'a requires that structures, systems, and components of nuclear power plants impo-tant to safety be protected against the effects of missiles that might result from equipment failures. Because flywheels have large masses and rotate at speeds of about 1200 revolutions per minute during normal reactor operation, a loss of integrity could result in high energy missiles and excessive vibration of the reactor coolant pump assembly. The safety consequences could be significant because of possible damage to the reactor coolant system, the containment, or the engineered safety features.

The potential for the reactor coolant pump flywheel to become a missile in the event of a rupture in the pump suction or discharge sections of reactor coolant system piping is under generic study by EPRI (Electrical Power Research Institute) and the NRC staff. EPRI has contracted Combustion Engineering, CREARE, and Massachussetts Institute of Technology to perform experimental and analytical work on two-phase flow reactor coolant pump performance. The pump test program is in progress and testing will be performed on a one-fifth-scale test loop at Combustion Engineering, and a 1/20-scale test loop at CREARE. The objective of the program will be, in part, to obtain empirical data to substantiate or modify current mathematical models used in predicting pump performance during a postulated loss-of-coolant accident. EPRI and its contractors plan to complete the program by late 1977.

5.2.7 Leakage Detection System

Coolant leakage within the primary containment may be an indication of a small through-wall flaw in the reactor coolant pressure boundary. The proposed systems for detection of coolant leakage to containment will provide: (1) diverse leak detec methods, (2) sufficient sensitivity to measure small leaks, (3) identification of the leakage source to the extent practicable, and (4) suitable control room alarms and readouts.

The primary method of detecting unidentified leakage into the containment will be by measurement of the sump level. Methods in addition to the sump level method will be the use of the containment particulate activity monitor and the containment gaseous activity monitor. Instrumentation will be provided to monitor identified leakage from the reactor vessel head closure seal, reactor coolant pump seal primary safety valves, and leakage through the steam generator tubes or tubesheet.

We find that the leakage detection system for detecting leakage from components and piping of the reactor coolant pressure boundary are in accordance with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." This provides reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions. We have concluded that compliance with the recommendations of Regulatory Guide 1.45 constitutes an acceptable basis for satisfying the requirements of Criterion 30 of the General Design Criteria, and that the proposed leakage control system meets the interface requirements stated in the CESSAR.

712 089

5.2.8 Inservice Inspection Program

Our evaluation of the inservice inspection program for systems within the nuclear steam supply system, which include all of the reactor coolant pressure boundary (ASME Code Class 1 components) and a part of the ASME Code Class 2 and Class 3 components, is presented in Section 5.2.8 of Appendix A to this report.

To ensure that during plant service lifetime no deleterious defects develop during service in ASME Code Class 2 system components, selected welds and weld heat-affected zones will be inspected prior to reactor startup and periodically throughout the life of the plant. In addition, Code Class 2 systems and Code Class 3 systems will receive visual inspections while the systems are pressurized in order to detect leakage, signs of mechanical or structural distress, and corrosion.

Engineered safety features, not part of Code Class 1 systems, represent an example of Code Class 2 systems. Examples of Code Class 3 systems are the component cooling water system and portions of the radwaste systems. All of these systems transport fluids. The applicant has stated that the design of Code Class 2 systems will meet the requirements of the ASME Code, Section XI. Compliance with the inservice inspections required by this code constitutes an acceptable basis for satisfying Criteria 36, 39, 42, and 45 of the General Design Criteria.

To ensure that all ASME Code Class 1, 2 and 3 pumps and valves will be in a state of operational readiness to perform necessary safety functions throughout the life of the plant, the applicant has committed to a test program which will include baseline preservice testing and periodic inservice testing. Such a program will provide for both functional testing of the components in the operating state and for visual inspection for leaks and other signs of distress.

The applicant has stated that the inservice test program for all Code Class 1, 2 and 3 pumps and valves will meet the requirements of Subsections IWP and IWV, respectively of the ASME Code, Section XI. Specific details of the testing program will be provided during the operating license review.

Compliance with the referenced code requirements constitutes an acceptable basis for satisfying the requirements of Criteria 37, 40, 43 and 46 of the General Design Criteria.

5.2.9 Loose Parts Monitor

The interface requirement stated in the CESSAR is that the balance of plant design include a loose parts monitor system having the capability to detect an impact of one-half foot pound or more on internal surfaces of reactor coolant system. The applicant has committed to install such a system but is still evaluating the extent of compliance with the sensitivity requirements. We conclude that the commitment to supply a loose parts monitor is acceptable at the construction permit stage of review. Additional discussion is included in Section 5.2.9 of Appendix A to this report.

5.5 Component and Subsystem Design

5.5.2 Steam Generator

Our evaluation of the steam generator is presented in Section 5.5.2 of Appendix A to this report. In addition to that evaluation, we have evaluated the factors that could affect the integrity of the steam generator tubes that will be used. We conclude that reasonable measures will be taken to ensure that the tubes will not be subjected to conditions that will cause deleterious wastage or cracking. Our conclusion is based on the following:

(1) The steam generators will be of advanced design with improved secondary water flow characteristics. This will provide more tolerance for occasional lack of control of the secondary water chemistry.

(2) All volatile treatment is planned for secondary water chemistry control, thereby minimizing the probability of deleterious local high concentrations of caustic or phosphate on the tubing.

(3) To further control impurities in the secondary water to very low levels, the proposed facilities will use condensate polishing.

(4) Access has been provided and provision has been made in the design for installation of equipment for the remote inservice volumetric inspection of steam generator tubes.

5.5.5 Residual Heat Removal System

For the residual heat removal system design the PSAR references the CESSAR design and satisfies the interface requirements for the balance-of-plant design. Our evaluation of the residual heat removal system design is presented in Section 5.5.5 of Appendix A to this report.

6.0 ENGINEERED SAFETY FEATURES

6.1 Design Considerations

Engineered safety features is the designation given to those systems which will be provided for the protection of the public and station personnel against the postulated release to the environment of radioactive products from the nuclear plant, particularly as the result of the loss-of-coolant accident. This section contains our evaluation of the engineered safety features which are within the balance of plant scope. Our evaluation of those engineered safety features which are within the scope of the standard reference system design is contained in Section 6.0 of Appendix A to this report.

Certain of these systems will have functions for normal plant operation, as well as serving as engineered safety features. Systems and components designated as engineered safety features will be designed to be capable of assuring safe shutdown of the reactor under the adverse conditions of the various postulated design basis accidents described in Section 15.0 of this report. Therefore, they will be designed to seismic Category I standards and must function even with assumed complete loss of offsite power. Components and systems will be provided in sufficient redundancy so that a single failure of any component or system will not result in the loss of the capability to achieve safe shutdown of the reactor. The instrumentation systems and emergency power systems will be designed for the same seismic and redundancy requirements as the systems they serve. These systems are described in Section 7.0 and 8.0 of this report.

6.2 Containment Systems

The containment systems for each unit will include a containment vessel, containment heat removal system, a containment isolation system, a combustible gas control system, and provisions for containment leakage testing.

6.2.1 Containment Functional Design

The containment for each unit will be a spherical steel vessel with a minimum net free volume of 3,300,000 cubic feet. The containment vessel will house the nuclear steam supply system, which will include the reactor, steam generators, reactor coolant pumps and pressurizer, as well as certain components of the plant's engineered safety feature systems. The containment will be designed for an internal pressure of 46.8 pounds per square inch gauge and a temperature of 280 degrees Fahrenheit.

712 092

The containment vessel will be completely enclosed by a seismic Category I shield building. The shield building will be a low leakage, reinforced concrete structure designed to provide biological shielding during normal operations and postulated loss-of-coolant accident conditions, and protection for the containment from atmospheric conditions and external missiles. The annulus between the containment and the shield building above the 92-foot elevation will serve as a space for collection and filtration of fission product leakage from the containment vessel. Following a loss-of-coolant accident the annulus ventilation system will circulate the annulus atmosphere through filters and will exhaust a portion of the circulating flow to the atmosphere at rates to achieve and maintain a reduced pressure in the annulus. The applicant has analyzed the containment pressure responses for postulated accidents in the following manner. Mass and energy release rates to the containment for postulated reactor coolant system pipe breaks were calculated by Combustion Engineering, Inc. in accordance with previously accepted methods presented in the CESSAR. The mass and energy release rates were then used as inputs to the Combustion Engineering CONTRANS digital computer code, which performs transient thermodynamic calculations with appropriate consideration of containment heat removal systems and structural heat sinks to calculate the containment pressure and temperature response.

The applicant has analyzed a number of reactor coolant system pipe break accidents including a spectrum of break locations and sizes. The postulated double-ended slot break at the pump suction of the reactor coolant system resulted in the highest calculated containment pressure which was about 43 pounds per square inch gauge. The loss of one of the two containment spray trains and full emergency core cooling system operation were conservatively assumed for the evaluation.

We have also independently analyzed the containment pressure response to a postulated double-ended slot break at the pump suction of the reactor coolant system using the CONTEMP1 computer code. Our analysis was based on the mass and energy release, containment structural heat sink, and spray system performance data provided by the applicant. Conservative condensing heat transfer coefficients to the structures inside the containment were also used. The results of our analysis confirm the acceptability of the peak pressure calculated by the applicant. We therefore conclude that the containment design pressure is acceptable since it provides a 10 percent margin above the peak calculated pressure.

The applicant has analyzed a spectrum of main steam line break accidents to determine the containment pressure and temperature response. The mass and energy release rate data used was based on previously accepted calculational methods described in the CESSAR. The applicant has also included in his analysis the volume of feedwater stored in the lines between the isolation valves and steam generator nozzles. At our request, the applicant reanalyzed the main steam line break accident but did not adequately justify the method of feedwater addition used for the zero power case. The CONTRANS computer code is used by the applicant to calculate the temperature and pressure response of the containment. The code has been modified such that the saturation temperature of the condensate rather than the containment vapor temperature is used to calculate the heat transfer from the containment atmosphere. We find this approach to be acceptable.

The applicant calculated the highest pressures and temperatures for the main steam line break accident occurring at power with an 85 percent break area. The applicant calculated a peak atmospheric temperature of 387 degrees Fahrenheit and a peak pressure less than the loss-of-coolant accident peak pressure.

We have done a confirmatory analysis using the CONTEMPT-24 computer code. Our analysis was based on the mass and energy release rate data provided in the CESSAR and the revised heat sinks presented in Amendment 20 to the PSAR. We have also accounted for the fluid volume in the feedwater lines by conservatively adding this volume of water to the containment before the containment spray system becomes effective and assuming the feedwater flashed to saturated steam in the steam generator.

We calculated a peak containment pressure that was less than the containment pressure for the design basis loss-of-coolant accident. The peak calculated vapor temperature is 403 degrees Fahrenheit, which is above the design temperature of 387 degrees Fahrenheit proposed by the applicant for essential safety-related instrumentation inside the containment. As discussed in Section 3.11 of this report the applicant in Amendment 27 increased the peak temperature for equipment environmental qualification to 407 degrees Fahrenheit which is in excess of an calculated temperature of 403 degrees Fahrenheit.

We have also examined the effects on the containment vessel and internals of the high temperatures resulting from main steam line break accidents. The results of our analysis indicate that the containment design temperature of 280 degrees Fahrenheit will not be exceeded.

The applicant has analyzed the pressure response within the various containment interior compartments, including the reactor activity, reactor cavity wall pipe penetrations, steam generator compartments and the pressurized skirt.

For the reactor cavity, reactor cavity pipe penetrations and steam generator compartments, the applicant postulated a single ended hot leg slot break (9.62 square feet) which is acceptably conservative. A double-ended guillotine rupture of the surge line was postulated in the pressurizer skirt. The CEFLASH-4A computer code was used to calculate the mass and energy release rates to the subcompartments. Combustion Engineering has made further conservative assumptions which act to maximize the mass and energy release rates to the compartments.

The Combustion Engineering DDIFF computer code was used by the applicant to calculate the subcompartment pressure responses. We have done a confirmatory analysis using the RELAP-3 computer code and our results are in good agreement with the applicant's

712 094

results. The applicant has committed that the design pressures for the subcompartments will be 40 percent greater than the peak pressures that will be calculated for the final design of the subcompartments.

During our review, the applicant reevaluated the consequences of inadvertent actuation of the containment spray system on the containment vessel. In the revised analysis, the applicant assumed that the containment atmosphere is initially at 120 degrees Fahrenheit, 14.7 pounds per square inch and 90 percent relative humidity, and that 590,000 gallons of water from the refueling water storage tank is added to the containment at a temperature of 80 degrees Fahrenheit. The applicant calculated a pressure drop of 1.93 pounds per square inch. We have done a confirmatory analysis and find that the containment external design pressure of two pounds per square inch gauge is acceptable provided there is a technical specification lower limit of 80 degrees Fahrenheit on the water stored in the refueling water tank.

We have completed our review of containment system functional design and conclude that the containment can be designed and built to function in accordance with the Commission's General Design Criteria including Criteria 16 and 50. We conclude that the functional design of the containment is acceptable for the construction permit stage of review.

6.2.2 Containment Heat Removal System

The containment spray system will be provided to remove heat from the containment following a postulated loss-of-coolant accident or a main steam line break accident. The spray system will consist of two spray trains and will be designed to accommodate any single failure and still be capable of supplying sufficient containment cooling to maintain the peak containment pressure below the design pressure. The containment spray system will serve only as an engineered safety feature and therefore will not be used for normal plant operation. It will be a seismic Category I system consisting of redundant piping, valves, pumps and spray headers. All active components of the containment spray system will be located outside of the containment vessel. Missile protection will be provided by direct shielding or physical separation of equipment. Redundant, completely separate sumps will be provided in the containment. In Amendment 28, the applicant in Table 1.7-1 of the PSAR has committed to design the sump screen assemblies in accordance with the provisions of Regulatory Guide 1.82, "Sumps For Emergency Core Cooling and Containment Spray Systems," to prevent debris from entering the spray system that could clog the spray nozzles.

The containment spray actuation signal will be generated by coincidence of a high-high containment pressure and a safety injection actuation signal, which will occur on high containment pressure. The spray pumps will initially take suction from the refueling water tank. When the water in the tank reaches a low level, the spray pump suction will be automatically transferred to the containment sump to initiate the spray recirculation phase. Operator action will be required to close the valves at the outlet of the refueling water tank.

We have reviewed the containment spray system and conclude that it is in conformance with Criteria 38, 39 and 40 of the General Design Criteria and with the recommendations of Regulatory Guide 1.1, "Net Positive Suction Head For Emergency Core Cooling and Containment Heat Removal System Pumps," and conclude that the containment spray system is acceptable for the construction permit stage of review.

6.2.3 Secondary Containment Functional Design

The secondary containment design will consist of an annular space between the containment vessel and the reactor, or shield, building above the 92-foot elevation. The annulus ventilation system will be designed to control the atmosphere in the annulus following a postulated loss-of-coolant accident and will maintain the annular space at a negative pressure of 0.5 inches water gauge. Plant areas that are contiguous to the containment vessel below the 92-foot elevation will not be included in the volume served by the annulus ventilation system. In these areas, the applicant proposes to provide leak chase channels over the containment vessel welds and over penetration welds and to vent the chases to the annular space above the 92-foot elevation. The applicant contends that a negative pressure will be maintained in the leak chase channels since they will be open to the annular space.

We have reviewed the functional design of the secondary containment system as modified during our review and the proposed periodic operability test program added to the PSAR during our review. We find that the modifications will assure that a negative pressure can be drawn in the secondary containment including both the annulus and the leak chase channels, and therefore, conclude that the functional design of the secondary containment system is acceptable and that the proposed testing will demonstrate its operability, including that of the leak chase channel network.

However, we were unable to conclude on the acceptability of the potential bypass leak paths identified by the applicant and the method of calculating the bypass leakage fraction. We requested additional justification by the applicant in accordance with the guidance in Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants." The applicant provided additional information in Amendment 28 to the PSAR. We completed our review of this information and by letter on February 22, 1977, requested additional information on six potential bypass leak paths.

Our evaluation of the additional information requested about potential bypass leak paths and calculated bypass leakage fraction will be presented in a supplement to this report. The applicant must provide justification for a bypass leakage fraction prior to a decision on issuance of construction permits. If that fraction is significantly higher than one percent as we have assumed in our dose calculations reported in Section 15 of this report, design changes may be required to decrease the calculated doses to acceptable levels.

712 096

6.2.4 Containment Isolation System

The containment isolation system will be designed to automatically isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of closed systems and isolation valves, will be provided to assure that no single active failure will result in the loss of containment integrity. The containment isolation provisions will be designed as seismic Category I equipment and will be protected against missiles which could be generated by the postulated design basis accidents.

The applicant will incorporate in his plant design the containment isolation provisions for certain system lines and the isolation signals which are described in the CESSAR. We have reviewed the interface requirements and conclude that they will be satisfied.

With regard to the isolation valves in the supply and exhaust lines of the containment purge system, the applicant has committed to keep the valves in the 42-inch lines closed during normal plant operation. The plant design also will include a containment pressure control system which will consist of a single, normally-open four-inch line and two 300 standard cubic feet per minute fans arranged in parallel. Provision will be made to filter the system exhaust. The four-inch line will also contain two automatic isolation valves in series, in accordance with Criterion 56 of the General Design Criteria. We have reviewed information provided by the applicant in Amendment 28 to justify that the system design will be consistent with Branch Technical Position CSB-4, "Containment Purging During Normal Plant Operations," We conclude that the information provided for normal power operation is acceptable. In our review of containment purge and pressure control systems, we identified a concern about a need for isolation activation signals derived from some parameter other than containment pressure. For the pressure control system, the applicant in Amendment 28 to the PSAR (Section 9.4.5.2.12) agreed to provide an engineered safety feature isolation signal to be designed to isolate the pressure control exhaust duct such that releases through that duct would be within the limits of Section 20.106 of 10 CFR Part 20.

The 42-inch lines would be open during refueling operations. In our accident analyses (Section 15.0) we assumed valves in these lines would close following a postulated refueling accident inside containment. However, the applicant has not described an isolation signal to close the large containment purge valves during refueling operations, nor did he completely describe other provisions for preventing release of radionuclides to the environment that would result in calculated doses in excess of a fraction of the guideline values of 10 CFR Part 100. We have discussed the need for this additional clarification with the applicant and will report the results of our review of that clarification in a supplement to this report.

Each isolation valve will be designed to permit periodic testing to verify valve operability and closure time. Design provisions will also be made to facilitate periodic local leakage rate testing of each isolation valve or barrier in accordance with the requirements of Appendix J to 10 CFR Part 50. We conclude that an acceptable isolation valve testing program can be developed using air or nitrogen as a pressurizing medium in lieu of using water as the applicant has proposed. Unless new bases are developed which we find acceptable we will require that periodic local leakage rate testing be accomplished without use of water as a pressurizing medium. This issue must be resolved prior to a decision on issuance of construction permits.

We have reviewed the containment isolation system for conformance to Criteria 54, 55, 56, and 57 of the General Design Criteria and Regulatory Guide 1.11 "Instrument Lines Penetrating Primary Reactor Containment." Subject to resolution of the matter regarding purging during refueling operations, we conclude that the applicant's proposed isolation system will be in conformance with Criteria 54, 55, 56 and 57 and Regulatory Guide 1.11, and therefore is acceptable.

6.2.5 Combustible Gas Control System

Following a loss-of-coolant accident, hydrogen may accumulate inside the containment as a result of (1) chemical reaction between the fuel rod cladding and the steam resulting from vaporization of emergency core cooling water, (2) corrosion of construction materials by the alkaline spray solution, and (3) radiolytic decomposition of the cooling water in the reactor core and the containment sump.

In order to mitigate the consequences of hydrogen accumulation in the containment, the applicant proposes to provide redundant hydrogen recombiner systems, which will be located outside containment, and a backup purge system. Each of the 100 percent capacity recombiners and the backup purge system will be capable of processing the containment atmosphere at a rate of 80 standard cubic feet per minute. The applicant will provide a capability to continuously monitor the hydrogen concentration within the containment following a loss-of-coolant accident.

The applicant has not made a final decision as to the type of hydrogen recombiner system that will be used. However, the recombiner system will incorporate several design features that are intended to assure the capability of the system to remain operable in the event of an accident. Among these are: (1) seismic Category I design , (2) protection from missile and jet impingement and (3) redundancy to the extent that no single component failure will disable both recombiners. At the operating license stage of our review, we will require that the applicant provide electric hydrogen recombiners of a type that we previously found acceptable, or other recombiners that we find acceptable in the interim period, or the applicant will be required to provide a complete description of the generic design of the hydrogen recombiner system selected and the test conducted to demonstrate the functional capability of the system.

The applicant's analysis of the post-loss-of-coolant accident production and accumulation of hydrogen within the containment is consistent with the guidelines of Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment

Following a Loss-of-Coolant Accident," including the assumptions of a five percent zirconium-water reaction in the reactor core. Our confirmatory analyses verify the acceptability of the hydrogen generation analysis provided by the applicant.

Our prior review experience for combustible gas control systems is that effective hydrogen control systems can be designed to conform to the requirements of Criteria 41, 42, and 43 of the General Design Criteria and Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident," and therefore, we conclude that an acceptable system can be provided for combustible gas control following a postulated loss-of-coolant accident. We will review the design of the system during the operating license stage of our review.

6.2.6 Containment Leakage Testing Program

The containment design will include the provisions and features to satisfy the testing requirements of Appendix J to 10 CFR Part 50. The design of the containment penetrations and isolation valves will permit periodic leakage rate testing at the pressure specified in Appendix J. Included are those penetrations that have gasketed seals and electrical penetrations.

The proposed reactor containment leakage testing program will comply with the requirements of Appendix J to 10 CFR Part 50. Such compliance provides adequate assurance that containment leaktight integrity can be verified throughout service lifetime and that the leakage rates will be periodically checked during service on a timely basis to maintain such leakages within the specified limits.

Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through leak paths will not be in excess of acceptable limits specified for the site. Compliance with the requirements of Appendix J constitutes an acceptable basis for satisfying the requirements of Criteria 52, 53, and 54 of the General Design Criteria.

6.2.7 Engineered Safety Features Air Filtration Systems

The engineered safety feature air filtration systems for each unit of the proposed facility will consist of process equipment and instrumentation to control the release of radioactive materials in gaseous effluents following a design basis accident.

Habitability Systems. There are two filtration systems designed for air cleanup in habitable areas. These are the control room ventilation system and the equipment and cable room ventilation system. Each unit will be provided with these two filtration systems.

(1) Control Room Ventilation System. The function of the control room ventilation system will be to supply air that is free of radioactive materials to the control

712 099

room after a postulated design basis accident and to pressurize the control room to a minimum of 0.05 inches water gauge. This system will permit operating personnel to remain in the control room following a postulated accident. The control room ventilation system will be a 100 percent redundant system, with each system having an intake design capacity of 1000 cubic feet per minute of air and a recirculating design capacity of 2000 cubic feet per minute of air. Each system will include (a) an electric heating coil, (b) prefilter, (c) high efficiency particulate air, (d) carbon adsorber, (e) a second high efficiency particulate air filter, and (f) a fan. The equipment and components will be designed as recommended in Branch Technical Position ETSB 11-1 (Rev.1), "Design Guidance for Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Reactor Plants," and will be located in a seismic Category I structure. Following a design basis accident, the pressurization and recirculation system will be automatically activated by a signal from radiation monitors, gas or smoke detectors located in the inlet ducts or be activated manually from the control room. Interconnections on the intake of this system with the equipment and cable room ventilation system intake provides 100 percent redundancy.

We have determined that the control room ventilation system will be designed in accordance with the guidelines of Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and is capable of maintaining a suitable control room environment following a design basis accident. We therefore find the system acceptable.

(2) Equipment and Cable Room Ventilation System. The function of the equipment and cable room ventilation system will be to supply air that is free of radioactive materials to these rooms after a design basis accident and to pressurize the rooms to a minimum of 0.05 inches water gauge. This system will provide a suitable environment for the operation of vital equipment during an accident. The equipment and cable room ventilation system is a 100 percent redundant system, with each system intake design capacity of 100 cubic feet per minute of air and a recirculating design capacity of 2000 cubic feet per minute of air. Each system will contain a set of components that is identical to the set of components provided for the control room ventilation system described above under (1), and which will also be located in a seismic Category I structure. Following a design basis accident, the pressurization and recirculation system will be automatically activated by a signal from radiation monitors, gas or smoke detectors located in the inlet ducts or be activated manually from the control room.

We have determined that the equipment and cable room ventilation system will be designed in accordance with the guidelines of Regulatory Guide 1.52 and will be capable of maintaining a suitable room environment following a design basis accident. We therefore find the system acceptable.

712 100

Controlled Access Area Ventilation Systems. There are three engineered safety features filtration systems designed to control the release of radioactive materials in gaseous effluents. These are the annulus ventilation system, the reactor building auxiliary equipment exhaust system and the fuel handling ventilation exhaust system. Each unit will be provided with all three of these filtration systems.

(1) Annulus Ventilation System. The function of the annulus ventilation system will be to produce and maintain a slightly negative pressure in the annular space between the containment and the reactor shield building in order to control the release of radioactive materials in gaseous effluents following a loss-of-coolant accident. The system will be activated by the containment high-high pressure signal. The annulus ventilation system will be a 100 percent redundant system. Each train will have a design capacity of 16,000 cubic feet per minute and will include (a) a demister, (b) an electric heating coil, (c) a prefilter, (d) a high efficiency particulate air filter, (e) a carbon adsorber, (f) a high efficiency particulate air filter, and (g) a fan. The equipment and components will be designed as recommended in our Branch Technical Position 11-2 (Rev. 1) "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Reactors Plants," and will be located in a seismic Category I structure.

We have determined that the annulus ventilation system will be designed in accordance with the guidelines of Regulatory Guide 1.52, and therefore, we find the system acceptable.

(2) Reactor Building Auxiliary Equipment Exhaust System. The function of the reactor building auxiliary equipment exhaust system will be to control the equipment space inside the reactor shield building at a slightly negative pressure with purge filtration to the unit vent thereby controlling the release of radioactive materials in gaseous effluents during normal operations, normal shutdown and following a loss-of-coolant accident. The reactor building auxiliary equipment exhaust system consists of four trains, each consisting of (a) a demister, (b) a electric heating coil, (c) a prefilter, (d) a high efficiency particulate air filter, (e) a carbon adsorber, and (f) a second a high efficiency particulate air filter. The average continuous purge rate will be 200 cubic feet per minute with the inlet supply automatically terminated following a loss-of-coolant accident. The equipment and components will be designed as recommended in our Branch Technical Position ETSB 11-1 (Rev. 1) and will be located in a seismic Category I structure.

We have determined that the reactor building auxiliary equipment exhaust system will be designed in accordance with the guidelines of Regulatory Guide 1.52, and therefore, we find the system acceptable.

(3) Fuel Handling Ventilation Exhaust System. The function of the fuel handling ventilation exhaust system will be to control the release of radioactive materials in gaseous effluents from the fuel handling area following a postulated fuel

handling accident. The system will be designed to maintain a slight negative pressure in the fuel handling area of the auxiliary building following a loss-of-coolant accident within containment, a fuel handling accident in the fuel handling building or loss-of-offsite power. The fuel handling ventilation exhaust system will be a 100 percent redundant system. Each train will have a design capacity for maintaining a minimum of 10 air changes per hour over the spent fuel pool and will include (a) a demister, (b) an electric heating coil, (c) a prefilter, (d) a high efficiency particulate air filter, (e) a carbon adsorber, (f) a second high efficiency particulate air filter, and (g) a fan. The equipment and components will be designed as recommended in our Branch Technical Position ETSB 11-1 (Rev. 1) and will be located in a seismic Category I structure. The fans will be operated during fuel handling and will automatically operate to maintain the pressure differential and radiation levels in the area.

We have determined that the fuel handling ventilation exhaust system will be designed in accordance with the guidelines of Regulatory Guide 1.52, and therefore, we find the system acceptable.

6.2.8 Containment Air Purification and Cleanup Systems

The containment spray system will be used for iodine removal following a postulated loss-of-coolant accident. The applicant has proposed to add trace quantities of hydrazine to the spray solution to enhance its elemental iodine removal effectiveness. We have previously reviewed the concept of hydrazine as a spray additive. This concept has been thoroughly investigated and its effectiveness demonstrated in small and large scale experiments, including test runs at the Containment Systems Experiment facility at the Pacific Northwest Laboratory.

The proposed system will employ the hydrazine additive in conjunction with crystalline disodium phosphate stored in the containment for post-accident control of sump hydrogen ion concentration (pH). This pH control agent will be stored below the post-accident sump water level in the containment, and will produce a sump solution in the range of hydrogen ion concentration (pH) of 7.0 to 7.3 when dissolved in the sump solution.

The hydrazine addition system will be designed to produce a spray additive concentration of 50 parts per million. Redundant additive pumps will be used to inject 35 weight percent hydrazine from the hydrazine storage tank to each train of the containment spray system.

Although post-accident circumstances are unlikely to require hydrazine injection beyond an initial four-hour period, the applicant will provide a 24-hour capacity tank. The excess capacity is sufficient for several additional injection periods, if the need should arise, because of decomposition of the hydrazine that was injected initially. We conclude that the proposed tank capacity is acceptable.

The trace quantity hydrazine additive has been shown to produce elemental iodine partition coefficients comparable to those achieved with sodium hydroxide systems. The iodine removal effectiveness of the system, therefore, will be mass transfer limited, and the iodine transfer across the gas film surrounding the spray drops may be expected to be the rate controlling process. The calculation models developed for predicting the effectiveness of sodium hydroxide spray systems, therefore, also apply to the hydrazine system. Our calculations indicate that, similar to most sodium hydroxide systems, the iodine removal rate constant for the hydrazine spray system exceeds the maximum value of 10 inverse hours, which represents the fastest rate process consistent with the iodine plate-out assumptions incorporated into the source terms of Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." The dose calculations of Section 15 of this report consequently, were performed using the conservative assumptions of iodine removal coefficients of 10, 0, and 0.6 inverse hours for the elemental, organic, and particulate forms of iodine, respectively, assuming an effective volume of 2,600,000 cubic feet which represents 79 percent of the total free volume of the containment. As for sodium hydroxide systems, the elemental iodine removal effectiveness of the spray system was assumed to diminish after the initial concentration for this form of iodine is reduced by a factor of 100.

We find that the proposed hydrazine addition system can provide the iodine removal coefficients used in our accident dose calculations in Section 15 of this report and conclude that it is acceptable for the construction permit stage of review.

6.3 Emergency Core Cooling System

Our evaluation of the emergency core cooling system is presented in Section 6.3 of Appendix A to this report.

The discussion in this section is limited to our review of the balance of plant design.

6.3.1 Design Basis

With regard to passive failures during long-term cooling, the applicant has stated in Section 6.3.1.5 of the PSAR that no limited leakage passive failure or the effects thereof such as flooding, spray impingement, temperature, pressure, radiation, or loss of net positive suction head in the emergency core cooling system during the recirculation mode will result in a loss of minimum acceptable system capability. Also, in the event of limited leakage passive failure in one subsystem during recirculation, the applicant has stated that appropriate personnel access provisions to the intact subsystem will be provided. We find these commitments to be acceptable for the construction permit stage of review. During the operating license stage of review, we will require and review more detailed aspects of the design regarding passive failures during long-term cooling.

6.3.3 Performance Evaluation

In response to the requirements of Section 50.46 of 10 CFR Part 50 regarding the evaluation of emergency core cooling system performance, the applicant has referenced the CESSAR evaluation of this system in the PSAR. In addition, the applicant has provided specific information required for the proposed facility design.

The staff issued a Safety Evaluation Report on the CESSAR in December 1975 (Appendix A to this report) in which it was concluded that the emergency core cooling system evaluation for the CESSAR was in compliance with the staff's requirements as stated in Section 50.46 of 10 CFR Part 50. The report noted that verification of certain design aspects regarding the safety injection tank isolation valves, the mini-flow bypass valves and the hot leg injection valves would be required during the final safety analysis review. The report also stated that each plant referencing the CESSAR would be required to submit plant specific information regarding the containment pressure calculations and submerged valves within containment during the construction permit review stage.

The applicant submitted plant specific containment parameters in Amendments 16, 20, and 21 to the PSAR which we conclude confirmed that the containment pressure calculations included in the CESSAR are conservative when applied to the proposed facility and are acceptable.

In Amendment 18 to the PSAR, the applicant stated that there were no valves inside the containment below elevation 79 feet that would be required to operate after the loss-of-coolant accident. The applicant in Amendment 29 will commit to environmentally qualify any valves, which during the final design review may be identified as valves that could be submerged. We find this commitment to be acceptable for the construction permit stage of review and will confirm documentation of this commitment in a supplement to this report.

In April 1976, Combustion Engineering informed the staff that an internal audit of the Combustion Engineering loss-of-coolant accident heatup code, STRIKIN-II, which had been used in the loss-of-coolant accident analysis for the proposed facility, had disclosed several errors in the code. After discussing the nature of the errors with the staff, Combustion Engineering made appropriate corrections to the STRIKIN-II code. At our request, Combustion Engineering also modified the STRIKIN-II code to comply with that part of the Commission's rule which states that return to nucleate boiling shall not be allowed in the model after critical heat flux is predicted. These modifications as well as the staff approved model change regarding the use of the FLECHT heat transfer coefficients during reflood, which are described in the Combustion Engineering report, CENPD-213, "Application of Reflood Heat Transfer Coefficients to C-E's 16 x 16 Fuel Bundle." January 1976, were incorporated in the reanalysis of the CESSAR system loss-of-coolant accidents. The results of calculations using the corrected and revised version of the STRIKIN-II code were documented by Combustion Engineering in CENPD-135, "STRIKIN-II-A Cylindrical Geometry

Fuel Rod Heat Transfer Program Supplement 4-P by Combustion Engineering," dated August 1976. We reviewed the modified STRIKIN-II code and found it acceptable. We documented our findings in the report, "Amendment 1 to the Status Report by the Directorate of Licensing in the matter of Combustion Engineering, Inc., ECCS Evaluation Model Conforming to 10 CFR 50, Appendix K," dated October 1976. The applicant referenced the above analysis as being applicable to the proposed facility in Amendment 28 to the PSAR.

The referenced analysis, as described in Combustion Engineering report CENPD-135, included a spectrum of seven breaks for the loss-of-coolant from major reactor coolant system pipe ruptures. The worst break was identified as the double-ended slot break located in the pump discharge and having a break discharge coefficient of 1.0.

Table 6.1 below summarizes the emergency core cooling system performance analysis for the limiting fuel rod at a linear heat generation rate of 13.4 kilowatts per foot and for the limiting break.

Table 6.1

EMERGENCY CORE COOLING SYSTEM PERFORMANCE

Parameter	Value	Criterion Limit
Peak Clad Temperature, Degrees Fahrenheit	2115	2200
Maximum Local Oxidation, Percent	16.8	17
Maximum Hydrogen Generation, Percent	0.525	1.0

As shown by Table 6.1, the predicted values for peak clad temperature, maximum local clad oxidation and maximum hydrogen generation are below the corresponding limits of 2200 degrees Fahrenheit, 17 percent and 1.0 percent respectively as specified in Section 50.46(b) of 10 CFR Part 50. The apparent improvement in performance over the previous calculation shown in Section 6.3.3 of Appendix A, which resulted in 12.1 kilowatts per foot as the limiting linear heat generation rate as compared to the present 13.4 kilowatts per foot limit, is due to the use of the new model for the FLECHT reflood heat transfer coefficients.

Based on our review, we conclude that the emergency core cooling system performance evaluation submitted for the proposed facility is in conformance with the peak clad temperature, maximum local oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling criteria stated in Section 50.46 of 10 CFR Part 50 with the provisions indicated for confirming those matters during the operating stage of review.

6.3.4 Tests and Inspections

For the tests and inspections to be performed for the emergency core cooling systems, the applicant references the CESSAR.

Our evaluation of the CESSAR preoperational tests of the system and component tests is presented in Section 6.3.4 of Appendix A to this report. Combustion Engineering has stated that the CESSAR system 80 tests would be performed in compliance with the Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors." The applicant references the CESSAR test description, but in Table 1.7.1 of the PSAR states that he intends only partial compliance with the provisions of Regulatory Guide 1.79. The applicant will provide clarification of his intent in Amendment 29 to the PSAR. We will require that he commit to conformance with the provisions of Regulatory Guide 1.79 which allow for justification of alternative test programs prior to a decision on issuance of construction permits. We will report the results of our review in a supplement to this report.

6.4 Engineered Safety Features Materials

The mechanical properties of materials selected for the engineered safety features within the balance of plant scope will satisfy Appendix I of Section III of the American Society of Mechanical Engineers (ASME) Code 1971 Edition, including the Winter 1973 Addenda, and the staff position that the yield strength of cold-worked stainless steels shall be less than 90,000 pounds per square inch.

We have reviewed the range of hydrogen ion concentration (pH) of the reactor containment sprays and the emergency core cooling water that will be maintained following a postulated loss-of-coolant accident. We find that the proposed control of hydrogen ion concentration will ensure freedom from stress-corrosion cracking of the austenitic stainless steel components and welds of the containment spray and emergency core cooling system throughout the duration of operation of these systems following a postulated loss-of-coolant accident. The controls on the use and fabrication of the austenitic stainless steel used in fabrication of the systems generally satisfy the recommendations of Regulatory Guide 1.31, "Control of Stainless Steel Welding," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices that will be performed in accordance with these requirements provide added assurance that stress-corrosion cracking will not occur during the postulated accident time interval. The control of the hydrogen ion concentration of the sprays and cooling water in conjunction with controls on selection of containment materials, in accordance with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident," provide assurance that the sprays and cooling water will not cause serious deterioration of the containment. The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on austenitic stainless steel components of the engineered safety feature system will be in accordance with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

Conformance with (1) the Codes and Regulatory Guide recommendations mentioned above, and (2) the staff positions regarding the allowable maximum yield strength of cold-worked austenitic stainless steel, and the minimum level of hydrogen ion concentration of the containment sprays and emergency core cooling water constitutes an acceptable basis for meeting the requirements of Criteria 35, 38, and 41 of the General Design Criteria.

In Amendment 28, the applicant identified some exceptions to Regulatory Guide 1.31 and 1.44 that he proposes to adopt. We will report the results of our review of this matter in a supplement to this report, and if significant exceptions remain, we will require that the applicant either conform with the recommendations of the guide or provide alternate acceptable bases for meeting the objectives of these guides prior to a decision on issuance of construction permits.

6.5 Control Room Habitability

The emergency protective provisions of the control room related to the accidental release of radioactivity or toxic gases are evaluated in this section. Relevant portions of the control room air conditioning system are discussed. Our evaluation of the control room air conditioning system is contained in Section 9.4.1 of this report.

6.5.1 Radiation Protection Provisions

The applicant proposed to meet Criterion 19, "Control Room," of the General Design Criteria by use of concrete shielding and by installing two separated fresh air inlets that will provide a source of fresh air for pressurization. In addition, the design will include redundant charcoal filter trains with each train designed to process 1000 cubic feet per minute of make-up air and 2000 cubic feet per minute of recirculated control room air. These trains will be automatically started upon detection of radiation.

Our review of the design of the system initially proposed by the applicant indicated a possibility of excessive operator doses following a postulated loss-of-coolant accident. The applicant reevaluated the design and submitted a modified system design in Amendment 20 of the PSAR. In the modified design outdoor air will be taken from two intake structures at widely separated locations. A radiation monitor will close intake dampers in one structure upon a high radiation signal. In the event of a high radiation signal in the air from both structures the control room will automatically be placed in a 100 percent recirculation mode. On the basis of our review we conclude that the modified system can be designed to perform in accordance with the requirements of Criterion 19 of the General Design Criteria, and therefore, is acceptable.

6.5.2 Toxic Gas Protective Provisions

We reviewed the design requirements for control room habitability following a postulated toxic gas release to ensure that operators can continue to carry out their monitoring and control functions. Our review was conducted according to the guidelines in Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a

Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

The applicant stated that no toxic gas will be stored on or in the vicinity of the site in sufficient quantity to pose a potential threat to the control room operators. Ammonia in a 30 percent aqueous solution will be stored in 13 gallon drums in the basement of the service building. Hydrazine will be stored in 55 gallon drums at the same location. These chemicals were determined to pose no threat to the control room operators in view of their low volatility and limited container size. The other hazardous chemicals discussed in Section 6.4 of the PSAR were also determined to be of no concern regarding control room habitability.

We have completed our review of the preliminary design and design criteria and conclude that acceptable provisions to protect against toxic gases can be provided. However, in Amendment 28, the applicant identified proposed exceptions to Regulatory Guides 1.78 and 1.95. We have reviewed the nature of the proposed exceptions and conclude that for the proposed facility the exceptions are not applicable. Hence, we did not review the acceptability of the exceptions which could be applicable to other licensed and proposed facilities of the applicant.

7.0 INSTRUMENTATION AND CONTROLS

7.1 General

7.1.1 Criteria

The Commission's General Design Criteria, the Institute of Electrical and Electronics Engineers Standards, including IEEE Std 279-1971 "Criteria for Protection Systems for Nuclear Power Stations," and applicable Regulatory Guides for Power Reactors have been utilized as the bases for evaluating the adequacy of the instrumentation and control systems.

7.1.2 Use of CESSAR in the Review

We concentrated our review of the instrumentation and controls systems on those aspects of the design that are not included in the CESSAR. Additionally, as reported in Appendix A to this report, during our review of the CESSAR, we identified several issues that were to be resolved by applicants referencing the CESSAR. During this present review, the applicant provided additional information to demonstrate that the proposed design will be in accordance with our additional requirements stated in Appendix A. These items are summarized below and are discussed further in the indicated sections of Appendix A.

(1) Interface Requirements

The staff's Safety Evaluation Report for CESSAR (Appendix A to this report) contains some interface requirements which are in addition to those contained in the CESSAR. We requested that the applicant commit to design the balance of plant to satisfy the CESSAR interface requirements and to satisfy the additional interface requirements identified by the staff in Appendix A (Sections 7.1, 7.2.3, 7.3.4, 7.4.2, and 8.1). In Amendment 26. the applicant stated, "It is the applicant's objective to meet these additional interfaces and, at this time, no major problems are foreseen in doing so. However, any exceptions to these additional interfaces will be addressed in the Project 81 FSAR." Because this commitment is in accordance with the Commission's policy on standardization, we find the commitment to be acceptable.

(2) Steam Line Break Isolation

In the CESSAR analysis of the steam line break accident inside the containment, credit is taken for the termination of main feedwater flow to the affected steam generator and the timely isolation of the intact steam generator.

These actions will be accomplished by the main steam isolation system which will effect the closure of the main steam and feedwater isolation valves.

With regard to the isolation of the intact steam generator at the steam side, our review as stated in Section 7.3.1 of Appendix A to this report, revealed two areas of concern in the event of a steam line break inside the containment. Resolution of these concerns requires that the applicant satisfy the additional interface requirements stated by the staff in Section 7.3.1 of Appendix A. The first of these two interface requirements is that the applicant commit to providing the necessary analysis with his operating license application if the analysis has not been completed by Combustion Engineering for the CESSAR and reviewed and approved by the staff, or to provide a design which meets the single failure criterion. In Amendment 26, the applicant made this commitment for all valves downstream of the main steam isolation valves but did not include any commitment for a design to preclude single failures of the electro-hydraulically-actuated atmospheric dump valves. In Amendment 28, the applicant changed the valve upstream of each of the dump valves from a normally open to a normally closed design. With this change, we conclude that the PSAR as amended through Amendment 28 provides an acceptable resolution of our concerns expressed in Section 7.3.1 of Appendix A to this report.

Appendix A to this report also included a requirement for a commitment to provide Class 1E signals to initiate the control systems for closing all of the valves downstream of the main steam line isolation valves if the analysis demonstrated that the blowdown of both steam generators is unacceptable. The applicant has stated that the turbine bypass systems will open at 1150 pounds per square inch gauge pressure. With regard to the turbine bypass system, we have noted that, given a single failure of a main steam line isolation valve, this system will close after the steam line header depressurizes to 1100 pounds per square inch gauge and is, therefore, acceptable. With regard to other valves, the applicant has committed to provide suitable design modifications if the analysis demonstrates that the blowdown of an additional steam generator through a particular valve is unacceptable. This commitment by the applicant is acceptable to us. Following documentation of this commitment in Amendment 29 to the PSAR, we will confirm in a supplement to this report that this matter is resolved.

(3) Safety Injection Tank Pressure Restoration

The design described in Section 7.6.5 of the CESSAR provides for the manual depressurization of the safety injection tanks to 400 pounds per square inch gauge during plant cooldown and for manual repressurization of the tank to 600 pounds per square inch gauge when the reactor coolant system pressure is being increased. The CESSAR also states that the administrative controls for the safety injection tank pressure change will be supplemented with an audible alarm to alert the operator of low tank pressure when the reactor coolant system pressure reaches 700 pounds per square inch gauge. The interface requirement includes a requirement that the applicant provide an alarm designed to meet the single failure criterion and be designed in accordance with the applicable Class IE requirements set forth in IEEE Std 279-1971 and IEE Std 308-1971.

An added interface requirement in Appendix A to this report is that instrumentation, which forms the basis for operator action, meet the requirements of IEEE Std 279-1971. Because the applicant has committed to the CESSAR interface criteria cited in Section 7.6.5 of Appendix A, we conclude that IEEE Std 279-1971 will be applied to all instrumentation which is required for safety, and that the applicant's commitment is acceptable for the construction permit stage of the review.

7.2 Reactor Trip System

Our review of the reactor trip system is presented in Section 7.2 of Appendix A to this report.

In Amendment 23, the applicant stated that in the event that the development program for the core protection calculator system described in the CESSAR proves to be unacceptable to us, an alternate design similar to those previously reviewed and approved by us will be implemented in the final design.

Section 7.2.1.1.2.4 of the CESSAR includes a requirement that the applicant's PSAR contain an evaluation of the CESSAR assumption that the maximum frequency decay rate at the reactor coolant pump buses will be three hertz per second. The analysis has not been provided in the PSAR but the applicant in the PSAR states that, "For any electrical system transients which could affect the departure from nucleate boiling ratio evaluation will be made to verify that these electrical systems transients will not cause departure from nucleate boiling ratio excursions below the minimum value of 1.3. The results of these evaluations will be contained in the Final Safety Analysis Report (FSAR). If it is determined by analysis that electrical system transients could effect the departure from nucleate boiling ratio, trips will be provided which are designed in accordance with the requirements of IEEE Std 279-1971, and any required equipment will be qualified for the safe shutdown earthquake and located in a Class I structure."

We have concluded that these commitments by the applicant are acceptable for the construction permit stage of the review.

7.3 Engineered Safety Features Actuation Systems

7.3.1 Introduction

The engineered safety feature actuation system and some of the systems actuated by that system are presented in CESSAR Section 7.3. Additional systems that are actuated by the engineered safety feature actuation system are discussed in Section 7.3 of the

PSAR. An example of one of these systems is the containment spray hydrazine addition system described in Section 7.3.6 that will be used by the applicant instead of the iodine removal systems described in the CESSAR.

Our evaluation of information pertaining to the single failure criterion is presented in Subsection 7.3.2 of this report and information pertaining to periodic testing is presented in Subsection 7.3.2. The remaining subsections present our evaluations of those engineered safety features actuation systems that were not included in the CESSAR.

7.3.2 Compliance with IEEE Standard 279-1971

Early in our construction permit review, we received an acceptable commitment to meet the requirements of Standard Review Plan Appendix 7-A Position 18, "Application of the Single Fa'. re Criteria to Manually-Controlled Electrically-Operated Valves." However, in Amendment 20, the applicant presented additional interface criteria which indicate that he differentiates between active and passive failures. Since this action appeared to contradict the earlier commitment, we requested clarification and in Amendment 23, the applicant stated, "Failures of electrical components such as motor-operated valves will be classified as single active component failures." Based on this clarification, we find that the applicant will provide systems designs which satisfy the requirements of Position 18 and, therefore, we conclude that the applicant has provided an acceptable commitment for assuring that the single failure criterion is met.

During our review, we requested that the applicant describe the design features and test programs that will be used to periodically test reactor protection and engineered safety features systems response times from sensor input to final actuator. The applicant has provided the following commitment in the PSAR:

"Provisions for reactor protection system and engineered safety features systems response time testing will be incorporated in the station design as referenced in CESSAR 7.2.1.1.9.8 and 7.2.1.1.9.7. These tests will include measurement of sensor response time to the extent practicable. Duke Power Company will follow the ongoing EPRI program involving system response time characteristics and response time measurements. Further description of the design provisions and identification and basis for any sensors which cannot be practically tested will be submitted in the FSAR."

We find this commitment to be acceptable for the construction permit stage of the review.

7.3.3 Auxiliary Feedwater System

Our evaluation of the instruments and controls for the auxiliary feedwater system is included in Section 10.5.2 of this report.

7.3.4 Annulus Ventilation System

The annulus ventilation system will be designed to circulate the atmosphere from the annulus surrounding the upper portion of the containment through filters for fission product removal. A negative pressure will be maintained in the annulus by exhausting flow to the environment at a rate equal to the sum of the containment leakage flow and the inward leakage flow inward through the surrounding shield building and upward through the containment support structure.

We found that the design criteria for the instrumentation and controls for this system satisfy our requirements identified in Section 7.1.1 of this report and are acceptable. However, the criteria presented in PSAR Section 7.3.2 were in contradiction with information provided in Table 6.2.3-2 and Figure 6.2.3-1 of the PSAR. This latter information indicated that the failure of a single sensor or single damper position control system for four dampers could result in an unacceptable annulus pressure and a loss of filtration by uncontrolled discharge to the unit vent. We also noted that failure of the hydrogen purge system valves located in a cross connection between the recirculation discharge ducts of the two annulus ventilation system trains could negate the independence of the trains.

In Amendment 26, the applicant provided a revised design in which eight dampers were arranged as two redundant pairs of control dampers in each of the two trains with separate control systems for each train. In Amendment 28, the applicant deleted the cross connection between the annulus ventilation and hydrogen purge systems.

On the basis of our review of the modified design we conclude that the design will satisfy the single failure criterion for instrumentation and controls and is acceptable.

7.3.5 Control Equipment and Cable Rooms Heating, Ventilation and Air Conditioning System

The control, equipment and cable rooms heating, ventilation and air conditioning systems will consist of two redundant 100-percent capacity systems. Each system will be designed to maintain the environment in each area they serve within acceptable limits for operation of plant controls. One train will always be in operation. The valves and dampers associated with both trains will automatically align for emergency operation and the train in operation will receive a start signal from the engineered safety features actuation logic described in the CESSAR so that the operating equipment will restart after a loss of offsite power. However, the applicant stated that the redundant system fans and chillers would not need to be automatically started. The applicant further stated that the failure of the operating fan and chiller to automatically restart constitutes an acceptable design because the operator has sufficient time and indication of failure to manually start the redundant train. The applicant had not provided the preliminary description of the instrumentation and the basis for the 160 minutes in which he claims the operator can

712 113

take action to prevent violation of the environmental limits for the safety equipment which is cooled by these systems.

Because it has been a staff position to require that redundant safety systems be actuated simultaneously, and because the applicant had not demonstrated that his design satisfies the single failure criterion, we concluded that the proposed design was not acceptable. In Amendment 23, the applicant committed to provide redundant and diverse Class IE sensors, indicators and alarms powered by separate Class IE power sources, to detect the failure of a heating ventilation and air conditioning system. The parameters to be detected are: (1) low chiller water flow in each train, (2) low air flow in ducting, (3) high fan discharge air temperature in each train, and (4) high temperature in the control room, equipment room and cable room.

We conclude for the heat loadings in the spaces to be monitored that the proposed sensors, indicators and alarms will assure sufficient time for the operator to take corrective action should the operating train fail to restart and that, therefore, an acceptable basis has been provided for reliance on operator action.

7.3.6 Containment Spray System

The containment spray hydrazine addition system proposed by the applicant will be activated upon receipt of a containment spray acutation signal. Our review of the proposed logic systems for the generation of the containment spray actuation signal shows that it is the same as the one described in the CESSAR which we have already reviewed and found acceptable as reported in Section 7.3 of Appendix A to this report.

7.3.7 Recirculation to the Refueling Water Tank

Section 7.3.4(2) of Appendix A to this report includes a requirement that the power connections to the isolation valves in the safety injection and spray pump recirculation lines to the refueling water tank satisfy the single failure criterion.

The applicant has proposed providing each of the two series valves with power from separate motor control centers in the same division. With maintenance of separate control and power wiring for each valve, the only single failure which could disable both valves would be a bus failure which would also disable the pumps.

Because these valves will be needed only when the associated pumps are operating, the failure of both valves to close will not present a safety hazard. The failure of a single valve to open will not constitute a hazard because redundant equipment is provided in a duplicate, electrically-independent and physically separated train.

We find that this design satisfies the single failure criterion and is, therefore, acceptable.

7.4 Systems Required for Safe Shutdown

In Section 1.9.3.3 of the PSAR, "Exceptions to CESSAR," the applicant does not state any exceptions to CESSAR Section 7.4, "Systems Required for Safe Shutdown." Therefore, in our review we have assumed that the results of our review reported in Section 7.4 of Appendix A to this report are applicable including the staff positions stated in Subsections 7.1.2.2 (Interfaces) and 7.1.2.3 (Steam Line Break). In addition, we informed the applicant that we require that the instrumentation which is used to monitor the cooling of the reactor coolant pumps meet the requirements for reactor protection systems (see Section 9.2.2 of this safety evaluation report) and that the description presented in PSAR Section 7.5 did not satisfy this requirement. In Amendment 28, the applicant modified PSAR Section 7.5 to read as follows, "Safety-related instrumentation will be incorporated to monitor component cooling service to the reactor coolant pumps. This instrumentation will be designed to assure that control room alarms are available upon loss of component cooling service. This design will be in accordance with applicable Section of IEEE 279-1971. However, it is noted that this design is more appropriately within the scope of IEEE P-466, "Criteria for the design of Safety Related Surveillance Instrumentation in Nuclear Power Generating Stations" presently under development. In regard to this standard, justification will be provided in the FSAR if IEEE P-466 is to be used in lieu of IEEE 279-1971." We find this commitment to be an acceptable basis for concluding that the plants will be adequately protected against the failure of the reactor coolant pumps from loss of cooling service.

7.5 Safety-Related Display Instrumention

In Section 7.5 of the PSAR, the applicant references CESSAR Section 7.5 and includes a discussion of design criteria for safety-related instrumentation not described in the CESSAR. On the basis of our review of this latter information we conclude that the safety-related display instrumentation will be designed to satisfy the criteria cited in Section 7.1.1 of this report and is acceptable.

7.6 Other Systems Required for Safety

7.6.1 References to the CESSAR

The applicant has referenced the CESSAR for the descriptions of the shutdown cooling interlocks, safety injection tank isolation valve interlocks, and refueling interlocks. Our position on the use of CESSAR is presented in Subsection 7.1.2 of this report (Subsection 7.1.2.2 "Interfaces", 7.1.2.3 "Steam Line Break" and 7.1.2.4 "Safety Injection Tank Pressure Restoration").

7.6.2 Equipment Provided by the Applicant

Our review shows that the systems and equipment which are not within the CESSAR scope but which are required for safety will satisfy all of the requirements of IEEE 279-1971. We conclude that the criteria for applicant's designs are acceptable. This conclusion is applicable to equipment described in Section 7.6 of the PSAR. During our review the applicant committed that equipment described in Section 7.6 would be subject to the same criteria as equipment described in Section 7-4 as equipment required for safe shutdown.

7.6.3 Class IE Diesel Lubrication System Instrumentation

The applicant proposed to use two oil pressure switches, in a two-out-of-two logic, which would meet all of the requirements of IEEE Std 279-1971 for protection of the diesels against a loss of lubrication. We find this concept acceptable. However, the applicant also proposed to check these sensors by cross checking between redundant counterparts. We did not find this acceptable because, IEEE 279-1971 requires that instruments used in cross checking bear a known relationship to each other and have read-outs available. Neither condition can be satisfied by a blind, go no-go, sensor such as a pressure switch.

The applicant was informed that diesel generator protective trips, with the exception of overspeed and high differential current, must be bypassed by an engineered safety feature actuation signal or meet all of the requirements of IEEE Std 279-1971. In Amendment 23, the applicant committed to test those sensors and to calibrate them in conformance with the calibration requirements of the technical specifications. We find this commitment acceptable for the construction permit stage of review.

7.7 Control Systems Not Required for Safety

In Amendment 23 to the PSAR, the applicant stated that in the event the development program for the core protection calculator system described in Sections 7.2.1 and 7.7 of the CESSAR proves to be unacceptable to the Commission, an alternate design similar to those previously reviewed and approved by the Commission will be implemented in the final design. This commitment, with regard to the reactor protection system, is acceptable for the construction permit stage of the review.

During our review, we did not find the applicant's statement in Section 7.7.2.3 of the PSAR that the instrumentation and controls of the instrument air system have no protection function to be acceptable. As discussed in Section 9.3.1 of this report, the applicant by letter dated February 8, 1977, provided clarification of the safety significance of this system.

7.8 Operating Control Systems

The applicant has committed to provide a bypassed status indicator in response to a CESSAR interface requirement and a commitment to satisfy the positions of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." We conclude that these commitments and the preliminary description of the proposed bypassed status indicator provided in Section 7.8 of the PSAR are acceptable at the construction permit stage of the review to assure that a acceptable system will be provided.

7.9 Technical Specifications

Appendix A to this report requires that the applicant address Branch Technical Positions 5, "Scram Breaker Test Requirements" and 9 "Definition and Use of Channel Calibration" which are found in Appendix 7A of the standard review plan. The applicant has stated that it is his intent to incorporate these requirements in the technical specifications. We conclude that this commitment is acceptable for the construction permit stage of the review.

8.0 ELECTRIC POWER

8.1 Introduction

Criteria 17 and 18 of the General Design Cr''eria, Regulatory Guides 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems"; 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies", 1.75, "Physical Independence of Electric Systems," and IEEE Std 308-1971, "Criteria for Class IE-Electric Systems for Nuclear Power Generating Stations," were utilized as the primary bases for evaluating the adequacy of the proposed electric power systems. In Section 1.9.3.3 of the PSAR, the applicant has committed to meeting all of the interface requirements of CESSAR Section 8.0, and in Section 8.0 of its PSAR has provided information in response to CESSAR interface requirements. This information reflects the need of an electric power system that will have two redundant and independent division arrangements for alternatirg current power and four redundant and independent division arrangements for direct current power. This is consistent with the required redundancy of safety-related components and systems included in the CESSAR.

We have identified in Table 8-1 of Appendix A to this report the interface acceptance criteria for the offsite and onsite power systems that we used as the bases for our review of the applicant's proposed design. These criteria include criteria in addition to the interface criteria in the CESSAR that we found were necessary for an assessment of the overall adequacy of the applicant's PSAR in combination with the CESSAR

8.2 Offsite Power System

8.2.1 Offsite Routing

The offsite alternating current power sources for the proposed facility will be provided by six double circuit 230 kilovolt lines, two of which will be located on each of three separate rights-of-way. All six circuits will be terminated in a 230 kilovolt breaker and a half switchyard.

8.2.2 Onsite Routing

The transmission lines will approach the switchyard on converging paths that will be located to maintain physical separation between the rights-of-way. Transmission lines will not cross each other and the horizontal distances between the rights-of-way will be greater than the tower heights. The interconnections between the switchyard and the unit generators and between the transformers and switchgear are described in the following sections.

712 118

In Amendment 28, the applicant provided the following criteria for the routing of power to remote structures, e.g., the nuclear service water pump structures:

(1) Class IE power and control cables routed outside the plant structure will be run underground.

(2) The underground Class IE cable system will consist of two concrete trenches containing cable trays. Redundant Class IE cables will be routed in separate trenches. The trenches will be equipped with drains and a concrete cover.

(3) The design will be such that any single design basis event applicable to the location of the cable trench will not result in the failure of redundant Class IE cables.

The proposed routing of the underground Class IE cable system in shown on Figure 3.2-1 of the PSAR.

As a result of these commitments and the criteria presented in the PSAR for the selection of cables, we conclude that remote structures will be served with adequate electrical power under all normal and abnormal events and, therefore, we find the design acceptable.

8.2.3 Switching Station

The electrical output from each unit will feed the two three-phase 230 kilovolt switching station mair ;uses through two half size feeders that will enter the switchyard at two separate bay locations. The two half sized step-up transformers, feeders and switchyard breakers bay will protect the integrity of each unit and the system against a single breaker, feeder or transformer failure.

The switchyard power circuit breakers will be operated by stored energy devices which will be charged from the switchyard 480 volt alternating current power system while the redundant protective relays and tripping circuits will be powered from redundant 125 volt direct current switchyard batteries. The two separate 480 and 277 volt alternating current power systems for the switchyard will normally be fed from the 6.9 kilovolt normal auxiliary power system will be supplied by the 6.9 kilovolt normal auxiliary power system of another unit. The switchyard auxiliary power systems will have redundant feeders to the switchyard load centers which will contain a step-down transformer and automatic transfer devices. The transmission network and the switchyards will be designed to maintain stable operation of the plant generators for single faults in the switchyard or transmission l s, and upon a sudden increase in system load or generation.

8.2.4 Unit Feeders

Each nuclear unit will generate power at 24 kilovolts and feed power through an isolated phase bus and two generator power circuit breakers to two independent half-size unit step-up transformers. One set of two unit auxiliary transformers, capable of supplying full capacity unit auxiliary power, will be connected to the isolated phase bus between each unit step-up transformer and its generator power circuit breakers. Each of the two independent step-up transformers will connect to the transmission system through its respective, independent circuit and switching station power circuit breakers.

Prior to and during start up of the nuclear units, the auxiliary power system will receive power from the offsite power system through the two independent circuits, the two independent step-up transformers, and the two independent sets of unit auxiliary transformers. During this period, the generator power circuit breakers will be open. The nuclear unit generator can be manually connected to the system by synchronizing across and then closing the generator power circuit breakers. The nuclear unit generator normally will be connected to the transmission system through two independent power transport circuits.

8.2.5 Bus Transfers

The 6900 volt normal auxiliary power system for each nuclear unit will consist of four switchgear groups. Each switchgear group will be connected through separate main breakers and buses to the two auxiliary transformers discussed in Section 8.2.4 above. The two auxiliary transformers will be energized by separate immediate access circuits. Normally, all four auxiliary transformers will be available, and each switchgear group will be energized by a single separate auxiliary transformer. In the event that one of the auxiliary transformers is lost, the switchgear breaker that is fed from the affected auxiliary transformer trips and, with breaker interlock devices, will initiate the closing of the alternate source breaker.

In the event that loss of one immediate access circuit de-energizes two auxiliary transformers in the same train, the two switchgear breakers that are fed from the affected independent circuit trip and, with breaker interlock devices, will initiate the closing of the respective alternate source breakers. The transfer of sources will occur within a maximum of eight cycles dead time and the two affected switchgear groups will then be energized from the two auxiliary transformers of the second immediate access circuit.

Normal bus transfers between the sources will be able to be initiated at the discretion of the operator from the control room. These transfers will be "live bus" transfers, i.e., the incoming source feeder circuit breaker will be closed onto the energized bus section and its interlocks will trip the outgoing source feeder breaker, which will result in transfers without power interruption.

Each unit will have two redundant and independent 4160 volt Class IE auxiliary power systems, identified as trains A and B, which normally will receive power from the 6900 volt normal auxiliary power system via the auxiliary transformers in the turbine building. The incoming source breakers will trip upon loss of normal power, and emergency power will be provided to each of the redundant 4160 volt Class IE auxiliary power systems trains by separate and completely independent diesel electric generating units.

8.2.6 Generator Power Circuit Breakers

The applicant has conducted load flow and transient stability analyses for the existing and proposed plants on his transmission system which indicate that the system remains stable over a wide variety of severe contingencies. The applicant has also made a commitment to provide a detailed study of load flow and transient stability in the FSAR which will demonstrate compliance of the preferred power system with Criterion 17 of the General Design Criteria. The applicant has also committed to provide generator circuit breakers which are identical in design to those to be used at Duke Power Company's McGuire and Catawba plants. The applicant also maintains that because the McGuire and Catawba generator power circuit breakers are being tested to 24 kilovolts no extrapolation of results is necessary. In the event that the McGuire and Catawba power circuit breaker qualification tests are not successful, the applicant has stated the system will be modified to satisfy Criterion 17 of the General Design Criteria. With regard to the commitments which the applicant has made for the qualification of the power circuit breaker, we note that these breakers are being tested and qualified for use at the Catawba and McGuire Stations. Because the model of the electrical grid which was used to determine the maximum fault current in this qualification program did not include the proposed Perkins and Cherokee Stations, it was not clear to us that the power circuit breaker will clear generator faults at Perkins, Cherokee, McGuire, or Catawba when the present distribution system is expanded to include all four stations. The applicant was informed of this and responded with the following commitment:

"The generator power circuit breakers are identical to the power circuit breakers used at both McGuire and Catawba Nuclear Stations for the same application. The generator power circuit breaker can receive fault current contributions from both the generator and the system during the worst case fault condition, hence the generator power circuit breaker ratings are selected to be compatible with the sum of the fault current contributions from both sources. The contribution from the system is subject to change in the event of future system changes; therefore, the power circuit breaker is rated to be compatible with the fault contribution calculated to be available from the system when all of the Perkins and Cherokee units are in operation plus a design margin to allow for future system changes. When system changes are contemplated, generator power circuit breakers and other equipment are analyzed for compatibility with the system requirements and duties. Equipment determined to be compatible is uprated or replaced as appropriate." This commitment is an acceptable basis to us for the construction permit stage of review.

8.2.7 Conclusions

Our review of the design of the power systems for the switchyards indicates that, if properly implemented, the offsite power system will satisfy the requirements of Criterion 17 of the General Design Criteria. However, the applicant's criterion, as presented, requires redundant alternating current and direct current sources for the proper operation of the offsite power supply and the implementation of these sources must be carefully reviewed as part of our operating license drawing review.

As a result of our review and in consideration of the applicant's commitments, we find that the 230 kilovolt sources on three rights-of-way will comply with the requirement of Criterion 17 for two physically-independent offsite power sources and that each unit feeder will be an immediate access line. We conclude, therefore, that an acceptable design will be provided.

8.3 Onsite Power Systems

The onsite alternating current power system for each unit will consist of two diesel generator sets, each exclusively feeding one of the engineered safety feature load groups. The two load groups are identified by the applicant as train A and B. Each diesel generator unit is operated independently of the other unit and, except during tests, will be disconnected from the utility power system. One of the two divisions in each unit will be able to supply sufficient power to provide for operation of sufficient safety features to cope with an accident or provide for a safe shutdown of the unit.

8.3.1 Alternating Current Power System

The diesel generator sets of each unit will have a continuous rating of approximately 6250 kilovolts each. The applicant has stated that the diesel generator sets will be tested to demonstrate compliance with Regulatory Guide 1.9 and to demonstrate a reliability of 0.99 at the 50 percent confidence level.

The applicant has stated that protective devices will be provided to protect the diesel generator against:

(1) low lube oil pressure

(2) engine overspeed

(3) high lube oil temperature

(4) high jacket water temperature

(5) low jacket water level

(6) low crankcase vacuum

(7) low jacket water pressure

(8) generator differential

As noted in Section 7.6.3 of this report, the low lube oil pressure, engine overspeed and generator differential protection permissive interlocks will be the only protective devices listed above that are not bypassed during starting of the diesel generators by an engineered safeguard signal. The applicant has stated that the low oil pressure switches used on the diesels as permissive interlocks will be periodically tested and calibrated in conformance with station technical test specifications, thereby ensuring their continuing reliable performance. In addition to periodically testing the low oil pressure switches, the applicant maintains that their reliability along with that of other components of the diesel generator unit will be established in the starting and reliability tests performed on the units. The diesel generators for each unit are housed in individual seismic Category I structures whose outer walls will be designed to withstand tornado wind loads and tornado missiles. The separation walls within the structures will be capable of withstanding internally-developed missiles.

The air starting system for each diesel gererator will have two receivers, each having a five start capacity. Because the performance of a diesel engine is very dependent upon its ability to obtain sufficient oxygen, the applicant will take all practical steps to include design features that will assure that the oxygen content of the engine's intake air will not become diluted by engine exhaust.

We conclude that an acceptable onsite alternating current power supply system can be provided that will satisfy the requirements of the General Design Criteria, IEEE Std 308-1971, IEEE Std 336-1974, IEEE Std 338-1971 and the recommendations of Regulatory Guides 1.6 and 1.9.

8.3.2 Vital Power Systems

Each nuclear unit will have four physically isolated 125 volt direct current vital power systems to supply 125 volt direct current and 120 volt alternating current to the instrumentation and control of the two independent engineered safety features load divisions. Each system will consist of a battery charger, powered by one of the two engineered safety features onsite power trains, a battery, a distribution center, a direct current power panel, a static inverter, an alternating current transfer switch and an alternating current power panel. The system will be designed with sufficient interlocks and manual disconnects so as to provide a high degree of electrical isolation between engineered safety features trains. However, for the proposed design, improper operation of maintenance breakers could, under certain fault conditions, compromise the independence of redundant engineered safety engineered safety features trains. The applicant initially proposed the use of administrative controls to prevent such compromises, but because of the CESSAR interface criteria in Amendment 11 agreed to provide interlocks in accordance with the recommendations of Regulatory Guide 1.5, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems."

The batteries will be sized to start and carry all direct current loads that are required to safely shut down and limit the consequences of a design basis accident and the loads of an adjacent board for a period of one hour. The batteries will be housed in separate battery rooms and each room will have separate ventilation to keep off gas below combustible concentrations. The applicant has proposed to perform a discharge capacity test in accordance with Section 5 of IEEE Std 450-1972 at intervals not to exceed 18 months.

The four batteries feed four static inverters which supply vital 120 volt alternating current loads. The alternating current distribution panels will be fed by these inverters through manual transfer switches which provide alternate power from two voltage regulators which will be fed from non-Class IE sources. We require interlocks that meet the requirement of Regulatory Guide 1.6 to preclude connection of redundant Class IE sources. In Amendment 23, the applicant presented revised direct current load distribution and interlocking schemes which satisfy the CESSAR interface criteria. Therefore, we conclude that the modified design is acceptable.

8.4 Physical Independence of Electrical Equipment and Circuits

The physical separation of electric circuits and equipment has been given early attention by the applicant. Redundant equipment has been assigned to different divisions with acceptable separation and/or barriers between these divisions.

The applicant states that the design for the facilities will conform to the recommendations of Regulatory Guide 1.75. This commitment satisfies the CESSAR interface requirements and, therefore, is acceptable.

712 124

9.0 AUXILIARY SYSTEMS

The auxiliary systems for the proposed facility are described in Section 9.0 of the PSAR. The auxiliary systems necessary for safe reactor operation or shutdown include the nuclear service water system, the component cooling water system for reactor auxiliaries, the ultimate heat sink, portions of the chemical and volume control system, the ventilation and air conditioning systems for the control building and engineered safety rooms, diesel generator fuel oil storage and transfer system, and the diesel generator auxiliary systems.

The systems necessary to assure safe handling of fuel and adequate cooling of the spent fuel include new and spent fuel storage facilities, the spent fuel pool cooling and cleanup system, the fuel handling facilities and a portion of the fuel handling system building ventilation system.

We have reviewed other auxiliary systems or portions of systems whose failure would not prevent safe shutdown, but could, either directly or indirectly, be a potential source of a radiological release to the environment. These other systems include the condensate storage facility, the demineralized water system, potable and sanitary water system, the low pressure service water system, recirculated cooling and ventilation cooling water system, the refueling and filtered water system, the compressed air system, the equipment and floor drainage system, the boron recovery system, and the equipment vent system. These systems will not be designed to seismic Category I requirements and have been reviewed and found acceptable. The acceptability of these systems was based on our review that determined that (1) at system interfaces or connections to seismic Category I systems or components, seismic Category I isolation valves will be provided to physically separate the non-essential portions from the essential system or component, and (2) the failure of non-seismic Category I systems or portions of the systems will not impair the function of safety-related systems or components located in close proximity.

The applicant has referenced the CESSAR for descriptions of the pressurizer relief tank, the fuel handling system, the chemical and volume control system, and the boron recycle system. Our evaluation of these is reported in Appendix A to this report.

Where systems or portions of systems are to be shared by the units of each facility, the applicant has stated that such sharing will not impair their ability to perform their safety functions. We have reviewed those systems and components to be shared, and find that they will be designed to meet the requirements of General Design Criterion 5, and that, therefore, the applicant's commitment is acceptable for the construction permit stage of the review.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

A new fuel storage pit will provide dry storage for approximately one-third of a core. The storage pit and racks will be designed to seismic Category I requirements. The racks will have a spacing that will be designed to maintain a maximum effective multiplication factor of 0.95 even in the event that the storage area is flooded with unborated water.

Based on our review, we have concluded that the design criteria and design bases for the new fuel storage facilities meet the requirements of General Criterion 62 of the General Design Criteria and the recommendations of Regulatory Guide 1.13, "Spent Fuel Storage Design Basis," including the recommendations on seismic design and missile protection, and are, therefore, acceptable.

9.1.2 Spent Fuel Storage

Spent fuel will be stored under water in the spent fuel storage pool. The seismic Category I spent fuel storage racks will be designed to prevent fuel assemblies from being placed in other than their prescribed locations. In Amendment 16, the applicant proposed a design change that would allow storage of two different spent fuel pool storage rack configurations. One of the proposed designs consists of an "open" spent fuel storage rack configuration that would accommodate an underwater storage of one and one-third cores of spent fuel (338 fuel assemblies). The second, a "canned" spent fuel storage rack configuration will be designed to accommodate the underwater storage of two and two-thirds cores of spent fuel (684 fuel assemblies). The open spent fuel racks will not have side enclosures and will allow the fuel pool cooling water to flow freely through the sides of the fuel racks. The design of the canned spent fuel racks will consist of stainless steel boxes of length equal to the length of a fuel assembly and are open at both the top and bottom ends. This will allow pool water to cool the spent fuel assemblies by natural convection. The open and canned fuel racks will have a nominal center-to-center fuel assembly spacing of 19 inches and 12.75 inches, respectively. We have determined that this spacing is sufficient to assure adequate cooling of the fuel (see Section 9.1.3). Analyses performed by the applicant and independently verified by us indicate that both types of storage racks will have a center-to-center spacing which is sufficient to maintain a maximum effective multiplication factor of 0.95 with a fuel assembly laying horizontally across the top of the storage racks and with the fuel pool flooded with unborated water.

The spent fuel pool will be designed to seismic Category I requirements and will be constructed of reinforced concrete with a stainless steel liner. Both the open and canned spent fuel storage racks will be designed to withstand the impact of a dropped spent fuel assembly from the maximum lift height of the spent fuel pool bridge hoist. The facility will be designed to prevent the cask handling crane from traveling over, or in the vicinity of the pool, thereby precluding damage to the stored spent fuel in the event of a dropped cask (See Section 9.1.4).

Based on our review, we have concluded that the design criteria and bases for the spent fuel storage facilities are in conformance with the requirements of Criterion 62 of the General Design Criteria and the recommendations of Regulatory Guide 1.13, including the recommendations regarding seismic design, missile protection and design compatibility with the handling of the fuel cask in the fuel pool areas, and are, therefore, acceptable.

9.1.3 Spent Fuel Pool Cooling and Cleanup Systems

The spent fuel cooling and cleanup systems will be designed to maintain the water quality and clarity of the pool water and to remove the decay heat generated by the stored spent fuel assemblies. The cooling system will be designed to seismic Category I requirements and will consist of redundant 100 percent capacity spent fuel pool cooling pumps, heat exchangers, and associated piping, valves and instrumentation. The capability to supply makup to the pool will be provided by permanently-installed seismic Category I connections from the service water system and the refueling water storage tank. Both of these sources will be designed to seismic Category I requirements. In addition, the spent fuel pool piping will be arranged so that the pool cannot be inadvertently drained to uncover the fuel. All lines that penetrate the pool will be equipped with siphon breakers.

The applicant has calculated the heat loads and fuel pool temperature resulting from (1) one-third and one and one-third cores of spent fuel stored in the pool using the open type fuel racks, and (2) two and two-thirds cores of spent fuel stored in the pool using the canned type fuel racks. We have independently calculated the heat loads for these storage conditions using Auxiliary and Power Conversion System Branch Technical Position APCSB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling." Our analysis results agree with the applicant's calculated heat loads for the one-third and two and two-thirds cores. The applicant's one and one-third core design heat load is based on the CESSAR decay heat values, which are acceptable.

The shutdown cooling system pumps will be able to be cross connected to the fuel pool cooling system. This cross connection will be made only when there is no fuel in the reactor vessel. For all normal storage conditions, the spent fuel pool cooling system will maintain the pool temperature at or below the design temperature of 130 degrees Fahrenheit. For the maximum storage conditions, the spent fuel pool cooling system, in combination with one train of the shutdown cooling system pumps and its associated shutdown heat exchanger system, or the shutdown cooling system itself will have sufficient heat removal capabilities to maintain the pool temperature at or below the design temperature of 130 degrees Fahrenheit. On the basis of our review which included an independent failure mode and effects analysis that indicated that the cooling system is designed to withstand the effects of a single active failure, we have concluded that the spent fuel cooling and cleanup system is acceptable.

The fuel pool cleanup system is a non-safety-related system and will be designed to non-seismic Category I requirements. The system pumps, piping and valves will be physically independent from the essential seismic Category I spent fuel pool cooling system and cross connections will not be provided. In addition, the failure of this system will not adversely affect any safety-related equipment.

Based on our review, we conclude that criteria and bases for the design of the spent fuel pool cooling and cleanup system are in conformance with Branch Technical Position APCSB 9-2 with respect to decay heat loads and the positions in Regulatory Guide 1.13, including the positions on availability of assured makeup sources and the requirements in Criterion 62 of the General Design Criteria and also meet the interface requirements of the CESSAR, and are, therefore, acceptable.

9.1.4 Fuel Handling System

The major portion of the fuel handling system, including the components required and the procedures for transferring fuel from the reactor to the spent fuel pool, is described in Section 9.1 of the CESSAR. Our evaluation of this portion of the fuel handling system is presented in Section 9.1.4 of Appendix A to this report. The equipment within the scope of the standard reference system design includes a refueling machine, fuel transfer system, control element assembly change platform, fuel handling tools, reactor vessel head lifting regulators, reactor internals handling equipment, spent fuel handling machines, new fuel elevator, underwater television system, dry sipping equipment, refueling pool seal, in-core instrumentation and control element assembly cutting tool, and a mechanism uncoupling tool. The new fuel handling area bridge crane, the new fuel area gantry crane and the spent fuel cask handling crane are within the scope of the balance of plant. Although the spent fuel handling machine is within the scope of the CESSAR, the applicant has taken an exception to the CESSAR by providing an alternate spent fuel handling machine which is identical to the refueling machine Appendix A to this report, we have found the design bases, system operation, safety evaluation and interface requirements for the refueling machine to be acceptable, and therefore conclude that the use of that design for the spent fuel handling application is acceptable.

The spent fuel cask loading pool area will be located adjacent to the spent fuel pool. The cask loading area will be separated from the fuel pool by reinforced concrete walls. Unacceptable damage to stored fuel due to a spent fuel cask drop will be prevented by limiting the travel of the spent fuel cask to an area which contains no safety-related equipment or stored fuel. The travel of the cask bridge crane is limited by mechanical stops and limit switches. Also, in Amendment 20, the applicant relocated the cask loading pool so that if the spent fuel cask were dropped, it would be prevented from rolling or toppling into the spent fuel pool by a physical barrier between the spent fuel pool and cask loading pool.

Based on our review of its design, including our previous review of the refueling machine, we conclude that the fuel handling system design criteria and bases are in conformance with the recommendations of Regulatory Guide 1.13, including the recommendation regarding protection of the spent fuel storage facility from the impact of unacceptable heavy loads carried by overhead cranes and also will be in conformance with the interface requirements, and are, therefore, acceptable.

9.2 Water Systems

9.2.1 Nuclear Service Water System

The nuclear service water system will provide cooling water to the safety-related plant systems for normal operation, for safe cold shutdown, and for the prevention and mitigation of postulated accidents. The portions of the nuclear service water system that will be contained within the auxiliary building will be identical for all three units.

The nuclear service water system for each unit will include a separate channel to supply cooling water to each of two redundant trains of essential equipment. For flexibility and diversity the channels will be cross connected at the supply and return headers to enable either channel to supply either of the essential equipment trains. Each cross connection will contain two normally-closed valves in series. Each essential equipment train will contain two component cooling heat exchangers, two component cooling pump motor coolers, one high pressure safety injection pump room air handling unit cooler, one low pressure safety injection pump room air handling unit air cooler, one containment spray pump room air handling unit cooler, one control, cable and equipment room air conditioning condenser, one diesel generator cooling water heat exchanger and one motor-driven auxiliary feedwater pump motor cooler. In addition, each supply header also will provide makeup water to the spent fuel pool and provide an emergency supply to the auxiliary feedwater pumps.

The nuclear service water system will include intake structures, pump structures, cooling towers, cooling tower makeup pumps, system pumps, piping, valves, instrumentation and controls.

Two separate and redundant nuclear service water system channels will supply cooling water to the respective nuclear service water trains of essential equipment for each unit as described above.

Each nuclear service water channel outside of the auxiliary building will include a separate and redundant intake structure, a pump structure, a cooling tower, a discharge structure, and supply and return piping.

The pump structure for one channel for each of the three units will include three separate pump compartments. The discharge from each pump will feed a corresponding channel in its respective unit. The cooling tower basin for each channel will be connected with the nuclear service water pump structure to permit bypassing the nuclear service water pond. This arrangment will allow two modes of operation for

712 129

the nuclear service water system: (1) cooling water supplied from and returned to the nuclear service water pond, and (2) cooling water supplied from the nuclear service water cooling tower basin and returned to the cooling tower. During the latter mode, normal makeup to the cooling towers will be provided by the cooling tower makeup system, which will be a shared system. An alternate source of makeup to all three units will be provided by two 100 percent capacity vertical pumps that take suction from the separate alternate nuclear service water pond.

Each essential equipment train will be designed to be capable of safely shutting down the unit. The nuclear service water system will be designed to seismic Category I requirements and to meet the single failure criterion. The system will be designed so that the environmental conditions resulting from postulated accidents will not impair the system's functional capability.

The primary mode of operation of the nuclear service water system will involve the use of nuclear service water cooling towers. Automatic transfer from a cooling tower to the nuclear service water pond will occur on low nuclear service water pump sump level, high nuclear service water supply temperature, low nuclear service water flow, or a loss of normal alternating current power. All the site-related structures and components of the nuclear service water will be designed to withstand a safe shutdown earthquake and to prevent any single failure from limiting the ability of the system to perform its intended safety function in the event of a postulated accident. The system will be protected against damage from postulated tornadoes and tornado missiles.

Based on our review, we conclude that the design criteria and bases for the nuclear service water systems meet the requirements of Criterion 44 of the General Design Criteria regarding their ability to transfer heat from safety-related components to the ultimate heat sink, and Criteria 45 and 46 regarding tests and inspections. We, therefore, conclude that the nuclear service water systems for the proposed facility will be acceptable.

9.2.2 Component Cooling System

The component cooling system will circulate cooling water in a closed loop to selected auxiliary components that will process potentially radioactive fluids. The component cooling system will be designed to function during normal, abnormal and accident conditions.

The component cooling system will consist of two redundant essential loops and two non-essential loops. Normal makeup to the system will be from the demineralized water storage system. Two identical component cooling pumps and two heat exchangers will be provided for each essential loop. Each pump and heat exchanger will be sized to provide the capability for transferring accident heat loads to the nuclear service water system.

During normal plant operation, one essential loop and both non-essential loops will be in operation. Upon a safety injection actuation signal the non-essential portion of the system will be automatically isolated and cooling water supplied only to the operating essential headers. In this mode, the component cooling system will provide the cooling water required under accident conditions for heat removal from the shutdown cooling heat exchanger, low pressure safety injection pump mechanical seal heat exchanger, high pressure safety injection pump mechanical seal heat exchanger, low pressure safety injection pump motor cooler, high pressure safety injection pump motor cooler, containment spray pump mechanical seal heat exchanger, containment spray pump motor cooler, fuel pool heat exchanger, and charging pump motor cooler. Flow to the letdown heat exchanger will be provided during normal operation by one of the essential loops but will be terminated during accident conditions.

The essential loops of the component cooling system, including the provisions for isolating the non-essential loops, will be designed to seismic Category I requirements. During an emergency condition each train of the component cooling system will be powered by an engineered safety features bus.

In the the proposed design the four reactor coolant pumps would be cooled by one of the non-essential loops of the component cooling water system. The seals and bearings of the reactor coolant pumps require continuous cooling for operation during normal plant operating conditions, anticipated transients, and following postulated accidents. As noted in Section 3.2.1 of this report and in Section 3.2.1 of Appendix A to this report, acceptance of these component cooling water lines to the reactor coolant pumps as Quality Group D designed to non-seismic Category I requirements is contingent upon favorable results of pump tests without component cooling water. Also, inadvertent failure or closure of any one containment isolation motor-operated valve in this portion of the non-essential loop would terminate the coolant flow to all of the pumps, an event that had not been analyzed by the applicant. We, therefore, requested the applicant to design this portion of the component cooling water system so that the following criteria are met:

(1) A single failure in the component cooling water system shall not result in fuel damage or damage to the reactor coolant system pressure boundary caused by an extended loss of cooling to the reactor coolant pumps. Single failure includes operator error, spurious actuation of motor-operated valves, and loss of component cooling water pumps.

(2) A moderate energy leakage crack or an accident that is initiated from a failure in the component cooling water system piping shall not result in excessive fuel damage or a breech of the reactor coolant system pressure boundary when an extended loss of cooling to the reactor coolant pumps occurs. A single active failure shall be considered when evaluating the consequences of this accident. Moderate leakage cracks should be determined in accordance with the guidelines of Branch Technical Position APCSB 3-1, "Protection Against Postulated Failures in a Fluid System Outside Containment."

To meet the two criteria above, the applicant, in Amendment 25, referenced Combustion Engineering's Topical Report CENPD-201, "System 80 Reactor Coolant Pump Performance."

This report included results of an analysis that indicated reactor coolant pumps described in the CESSAR could potentially operate with loss of cooling for longer than 30 minutes without loss of function or the need for operator corrective action. As a result of our review of CENPD-201, we concluded as reported in Appendix A to this report that Combustion Engineering will need to perform a reactor coolant pump test to verify the results of the analysis presented in Appendix A of CENPD-201. We also concluded that safety grade instrumentation would be required in order to assure a period of 20 minutes within which the operator would have sufficient time to initiate manual protection of the plant. As stated in Section 7.4 of this report, the applicant in Amendment 28 modified the PSAR to satisfy this requirement.

Alternatively, if it cannot be demonstrated by the necessary pump testing that the reactor coolant pumps will operate for more than 30 minutes without loss of function or operator corrective action, the applicant for the proposed facility shall in the FSAR:

(1) Provide safety grade instrumentation consistent with the criteria for the protection system to initiate automatic protection of the plant. For this case, the component cooling water supply to the seal and bearing of the pump may be designed to non-seismic Category I requirements and Quality Group D, or

(2) Design the component cooling water supply to the pumps to be capable of withstanding a single active failure or a moderate energy line crack as defined in Branch Technical Position APCSB 3-1 and to seismic Category I, Quality Group C and ASME Secton III, Class 3 requirements.

Based on our review, and subject to successful testing of reactor coolant pumps for 30 minutes without component cooling, we conclude that the component cooling system design criteria and bases are in conformance with the requirements of Criterion 44 of the General Design Criteria regarding the ability to transfer heat from safety-related components to the ultimate heat sink under normal and accident conditions and to meet the single failure criterion. We further conclude that the system design criteria and bases meet the requirements of Criteria 45 and 46 regarding system design that allows performance of periodic inspections and tests, including functional testing and confirmation of heat transfer capabilities. The component cooling, system design criteria and bases also meet the CESSAR system interface requirements.

9.2.5 Ultimate Heat Sink

The ultimate heat sink will be designed to provide a reliable source of cooling water for safe shutdown of the reactors following a postulated accident. Additional information on the ultimate heat sink is provided in Sections 2.4 and 9.2.1 of this report.

The ultimate heat sink for the three units will consist of a nuclear service water pond and two separate and redundant mechanical draft cooling towers and a source of makeup water for these towers. Each cooling tower and each pond will be capable of dissipating the heat load generated as a result of the postulated accident condition. The nuclear service water pond and associated intake and pump structures will be designed (1) to withstand the safe shutdown earthquake, and (2) to the single failure criterion such that a single failure will not impair their functional capability. In Amendment 20, the applicant states that the cooling towers will not be required following a postulated accident condition, and therefore, they will not be designed to seismic Category I requirements.

The applicant's analysis for the ultimate heat sink capability is based on the assumption that one unit experiences a loss-of-coolant accident and the other two units are capable of normal shutdown. For this condition, the nuclear service water would be recirculated to the pond for a period of 30 days. The pond will be sized for this condition assuming that no makeup water is available. The applicant has submitted values for the heat rate and total integrated heat rejected due to decay heat, rejected heat from station auxiliary systems and containment sensible heat. We have reviewed these values and find them acceptable.

Based on these heat inputs and conservative meteorology, the applicant has calculated that the nuclear service water pond will contain enough water and heat dissipation capability to maintain both units in safe shutdown for a period of 30 days. Assuming the most conservative recorded 30-day period, the maximum intake temperature was calculated by the applicant to be less than 100 degrees Fahrenheit for the safe shutdown of two units, and a loss-of-coolant accident in the other unit. Assuming seepage losses, evaporation, and drawdown, the applicant calculated the total water loss for the 30-day period to be within acceptable limits. We have performed independent analyses, and concur with the applicant's results. We conclude that the nuclear service water pond volume and heat dissipation capabilities are in accordance with Position 1 of Regulatory Guide 1.27, "Ultimate Heat Sink," and are, therefore, acceptable.

The applicant has further demonstrated to our satisfaction that the ultimate heat sink will be designed in accordance with Position 2 of Regulatory Guide 1.27, namely with the capability to withstand each of the most severe natural phenomena expected or a single failure of man-made structural features.

Based on our review, we conclude that the ultimate heat sink design criteria and bases are compatible with the positions of Regulatory Guide 1.27, and are, therefore, acceptable.

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

As noted in Section 7.7 of this report, the applicant stated in Section 7.7.2.3 of the PSAR that the instrumentation and controls of the instrument air system have no protective function. By letter dated February 8, 1977, the applicant has described the safety functions of the instrument air system and committed to preoperational

712 133

testing of the instrument air system in accordance with Regulatory Guide 1.80, "Pre-operational Testing of Instrument Air Systems" except for two tests not applicable to the proposed design. We conclude that this commitment which will also be documented in Amendment 29 is acceptable at the construction permit stage of review.

9.4 Air Conditioning, Heating, Cooling and Ventilation Systems

The safety-related ventilation systems that are required for equipment cooling include the control, equipment, and cable room ventilation system, the fuel handling ventilation portion of the auxiliary building ventilation system and the diesel building ventilating system. The function of these ventilation systems will be to maintain these areas within the thermal and air quality limits required for operation of plant controls and uninterrupted safe occupancy of required manned areas during normal operation, shutdown and post-accident conditions. The applicant initially stated that the reactor building auxiliary equipment ventilation system would not be required to mitigate the consequences of an accident within the reactor building. We did not agree with this statement, and as reported in Section 9.4.7 the applicant provided changes in his design criteria to make them acceptable.

The applicant maintains that ventilation systems to be provided for areas that contain safety-related equipment, including the emergency core cooling system pump rooms, containment heat removal system pump rooms and the nuclear service water system pump house, will maintain acceptable environmental conditions. However, at our request the applicant, in Amendment 11, agreed to provide supplemental seismic Category I coolers unless results of analyses at the operating license stage of review demonstrate that supplemental cooling is not required.

9.4.1 Control, Equipment and Cable Rooms Ventilation System

The control, equipment and cable room heating, ventilation and air conditioning system will be designed to maintain the temperature of the control, equipment and cable room areas within the limits required for operation of plant controls and uninterrupted safe occupancy during post-accident conditions.

Two 100 percent capacity seismic Category I chilled water systems will be provided to remove heat from the control, equipment and cable rooms air conditioning system. These systems will be designed to maintain the environment at a temperature of 75 degrees Fahrenheit during normal operation and a maximum of 90 degrees Fahrenheit during accident conditions. The nuclear service water system will remove the heat from this chilled water system.

Based on our evaluation and failure analyses, we have determined that the design of the air conditioning and ventilation system for the control equipment and cable rooms contains sufficient component redundancy and physical separation and meets the single failure criterion so that air conditioning and ventilation will be assured when required for anticipated operating conditions.

Based on our review, we conclude that the system design criteria and bases are in conformance with the requirements of Criterion 19 of the General Design Criteria with regard to the capability to operate the plant from the control room during normal and accident conditions, and the applicable positions set forth in Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and, therefore, are acceptable.

9.4.2 Fuel Handling Area Ventilation System

The fuel handling area ventilation system will be designed to provide ventilation, heating and cooling for the fuel handling area and to maintain a suitable environment for personnel and vital equipment during a postulated accident. The system will consist of a fuel handling ventilation supply system and fuel handling ventilation exhaust system.

The supply system will include one 100 percent capacity supply fan, heating and cooling coils. The exhaust system will consist of two separate 100 percent capacity exhaust trains, each with its own intake from the fuel handling area, filter train, exhaust fan, and ductwork. The filter train for each exhaust train will include prefilters, absolute filters and charcoal filters. The system will be capable of providing ten air changes per hour over the fuel pool to continuously purge the area. The exhaust system which will be operating whenever fuel handling operations are in progress will be designed to maintain a negative pressure to prevent outleakage from the fuel handling area in the event of a fuel handling accident.

In Amendment 20, the applicant stated that the entire fuel handling area exhaust system, which includes ductwork, dampers, filters and exhaust fans, will be designed as an engineered safety feature system and, therefore, will meet seismic Category I requirements.

Based on our review, we conclude that the design criteria and bases for the fuel building ventilation system meet the position set forth in Regulatory Guides 1.13 and 1.52 and, therefore, are acceptable.

9.4.6 Diesel Building Ventilation System

The diesel building ventilation system will be designed to provide a suitable environment for the operation of equipment and access of personnel for inspection, testing, and maintenance. Heating will be provided by electric coils and cooling will be provided by outside air. The diesel building inside temperature will be automatically maintained between a temperature of 55 and of 95 degrees Fahrenheit when the engine is shut down and less than 125 degrees Fahrenheit when the engine is in operation.

The system for each diesel generator will consist of one 100 percent capacity ventilation fan, two 50 percent capacity ventilation fans, ductwork, dampers, heaters and an air filter. The 100 percent ventilation fans will be designed to maintain the building temperature within acceptable environmental limits when the diesel engine is shutdown. The diesel combustion air will be supplied from the same intakes that supply the 100 percent capacity fan. When the diesel engine starts the fan inlet damper will automatically close to preclude the possibility of room air circulation to the combustion air intake in the event of a fan failure.

The two 50 percent capacity ventilation fans will be designed to maintain the minimum ventilation requirement of the diesel building during diesel engine operation. A separate ventilation system will be provided for each diesel generator; therefore, a single failure in one system would not prevent the redundant diesel from performing its safety function.

The diesel building ventilation system will be designed to the requirements of seismic Category I. The air intake and discharge will be protected against tornadoes and tornado missiles.

We have reviewed the diesel building ventilation system design criteria and bases and have found them to be acceptable.

9.4.7 Reactor Building Auxiliary Equipment Ventilation System

The reactor building auxiliary equipment ventilation system will be designed to provide a thermal environment within acceptable design limits with respect to personnel and operating equipment, to maintain the building under negative pressure, and to control releases during plant operation and following a postulated loss-of-coolant accident.

The system will consist of two redundant, physically separated, independent supply and exhaust ventilation system trains. Each system train will contain a 100 percent capacity supply fan, two 100 percent capacity exhaust fans, cooling coils, prefilters, preheaters, absolute filters and carbon filters. Following a postulated loss-of-coolant accident, each of the exhaust system trains will provide ventilation to those areas in the reactor building that contain redundant components of safety-related systems. This portion of the ventilation system is safety related and is designed (1) to seismic Category I requirements, and (2) to meet the single failure criterion.

The exhaust system will have a greater capacity than the supply system in order to maintain the ventilated spaces at a slight negative pressure during normal operations and shutdown. Following a postulated loss-of-coolant accident, the exhaust system will operate and the supply system will be shut off to assure that a large negative pressure will be maintained so that any radioactive leakage will be processed through the carbon filters enroute to its release to the atmosphere. Additional information pertaining to the effect of containment bypass leakage on the system is presented in Section 6.2.3 of this report.

In Amendment 25 in response to our request, the applicant has agreed to provide safety grade isolation dampers at the reactor building auxiliary equipment ventilation systems boundary so that the negative pressure and leakage control can be maintained in the event of a failure in one train of the ventilation system.

Based on our review, we conclude that the design criteria and bases for the reactor building auxiliary equipment ventilation system meet the recommendations set forth in Regulatory Guide 1.52, and therefore, are acceptable.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

During our review of the fire protection system, we requested that the applicant conduct a reevaluation of the proposed fire protection provisions and that he compare those provisions, in detail with the guidelines in Appendix A to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Plants Docketed Prior to July 1, 1976."

By letter of September 30, 1976, we requested additional information from the applicant on the implementation of these guidelines. By letter dated October 29, 1976, the applicant stated that the information requested would be provided by September 1, 1977.

On the basis of the applicant's commitments to meet the requirements of the Commission's General Design Criteria and to provide the additional information in accordance with our new guidelines, we conclude that the application is acceptable with respect to fire protection for the construction permit stage of review. It is our intent to provide the applicant with the results of our review of the information he plans to submit by September 1, 1977 on a timely basis such that he can effectively implement the guidelines during the evolution of the final design. We conclude that these measures provide sufficient assurance for the construction permit stage of our review that the applicant can provide a fire protection system that meets our requirements.

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

The diesel generator fuel oil storage and transfer system will be designed to provide fuel oil storage and transfer capability to allow operation of each emergency diesel generator for seven days.

This system will consist of two separate and independent trains, one for each diesel generator in each unit at each site. A cross connection with two normally-closed valves on the suction side of the fuel oil transfer pumps will permit transfer of oil to either diesel engine from either diesel fuel oil storage tank. Each system also will include a day tank which will hold a one hour supply of fuel oil. The diesel generator fuel oil storage and transfer system will be designed to seismic Category I requirements. Each diesel generator and the fuel oil storage and transfer system for that diesel will be housed in a separate diesel generator building. This building also will be designed to seismic Category I requirements and will be tornado missile and flood protected.

The fuel oil transfer pumps will be powered from separate emergency buses. Based on our independent evaluation, we have determined that the integrated design of the diesel generator fuel oil storage and transfer system satisfactorily will meet our single failure criterion.

Based on our review, we conclude that the diesel generator fuel oil storage and transfer system design criteria and bases meet the CESSAR interface requirements, and that the system has adequate capacity and can perform its designated safety functions and is, therefore, acceptable.

9.5.5 Diesel Generator Auxiliary Systems

The diesel generator auxiliary systems will consist of the diesel generator cooling water system, diesel generator starting system and the diesel generator lubrication system. The diesel generator closed cooling water system will be designed to maintain the temperature of the diesel engine within a safe operating range and to reject heat to the nuclear service water system. The system will be designed to seismic Category I requirements.

Each diesel generator will be provided with two separate and independent compressed air starting trains. By letter dated February 8, 1977, the applicant committed that each train will be designed to be capable of providing five starts including air compressor and starting air tanks. The starting air system will be designed to seismic Category I requirements.

Each diesel generator will be provided with a lubrication system designed to supply lubricating oil to the diesel generator system. The system will be designed to seismic Category I requirements.

We conclude that the diesel generator auxiliary systems design criteria and bases meet their designated safety functions, have the needed capacity and will, therefore, be acceptable.

9.5.6 Storage of Compressed Gases

The storage of containers of gases under pressure such as nitrogen, hydrogen, oxygen, compressed air, and carbon dioxide tanks, was reviewed, since these gases will be used in the operation of the power plants.

The applicant has considered these potential missile sources since a failure could affect safety-related systems or components. Measures within the facility to provide protection against potential missiles include: (1) provision of relief valves on tank with set points below the design pressures of the tanks; (2) location of tanks in limited access areas that are physically separated from safety-related components; (3) anchorage of tanks and cylinders so that they will not become missiles themselves following the failure of attached piping; (4) design of supply lines located in seismic Category I structures or near safety-related components to prevent damage by dynamic pipe movement and/or rupture; and (5) the location of gas storage facilities in relation to equipment essential for initiating and maintaining a safe reactor shutdown to preclude the possibility of interaction in the event of an incident.

Based on our review, we conclude that the design criteria and bases for the storage of compressed gases will preclude adverse effects on safety-related components and are, therefore, acceptable.

9.5.8 Permanent Dewatering System

The permanent dewatering system is a safety-related system that will consist of an underdrain system, an exterior wall drain system, a dewatering sump in the auxiliary building, and a two-train dewatering sump pump system with associated valves, instrumentation, and piping. All structures and components of the dewatering system will be designed to seismic Category I requirements and the system will be designed to pump normal groundwater flows from the sump while maintaining the groundwater level below the foundation elevation of the shield building and the auxiliary building. A separate dewatering system will be provided for each unit. Additional information on the permanent dewatering subsystems design can be found in Section 2.4.5 of this report.

Permanent dewatering system redundancy, from the sump pumps to the point where each discharge line discharges to the yard storm drain system, will be provided to meet the single failure criterion. In response to our request, the applicant in Amendment 27 agreed to provide one spare sump pump for the three units on the site. Power for the permanent dewatering system will be provided from the Class IE auxiliary power system. Emergency power will be provided from an associated emergency diesel generator. The two 120 gallon per minute seismic Category I pumps will be automatically started to maintain normal water level in the sump. Based on an estimated maximum groundwater inflow rate and with one pump running and a 15x15-foot sump cross section, a 1.25 foot per hour rate of water rise could occur in the groundwater drainage sump. A five-foot normal water level operating range will be used. In the event a pump fails to start, an alarm would be annunciated in the control room and when the water rises above the normal operating level, the second pump would be automatically started. A high level alarm will also be provided to alert the reactor operator. If both pumps should fail to start and the water rises in the sump, there is an additional 6.7 feet above the alarmed high water level for water rise within the sump that would allow 5.4 hours for corrective action before overtopping of the sump begins. We have concluded that the 5.4 hours is adequate to restore pumping capability. On this basis, we conclude that the permanent dewatering sump size and the sump pump capacity are acceptable for the estimated groundwater flow rates.

The capacity of the permanent dewatering system for each unit was based on the largest calculated normal groundwater inflow condition of 35 gallons per minute. A postulated moderate energy leakage crack in a piping system where it penetrates the exterior wall drain system could produce an additional flow into the system. Therefore, the applicant at our request analyzed the consequences of some of the events which would cause the flooding of the permanent dewatering system. In Amendment 27, the applicant reported on the result of his analysis of two assumed pipe breaks, the failure of the condenser circulating water system pipe in the turbine building or in the powerhouse yard and the failure of a nuclear service water system pipe in the wall drain system. The analyses of flooding of the permanent dewatering system due to failure of the condenser circulating water system pipe is given in Section 2.4.5 of this report.

The postulated failure of the nuclear service water pipe would result in the highest flooding rate calculated by the applicant. In the analysis a leakage crack in the pipe was postulated to occur within the wall drain system. This assumption is conservative because the leakage crack must occur in a specific section that is small compared to the total length of approximately 1250 feet of the nuclear service water system piping. However, in the event that a leakage crack would occur at this location, approximately 90,000 gallons of water could be added to the permanent dewatering system. This total includes the water pumped for a period of 30 minutes before the isolation valve is closed and also the trapped water in the pipe between the isolation valve and the leakage crack. The analysis shows that for this condition, overtopping of the sump would occur. The overflow will be dispersed on the auxiliary building floor to a depth of 2.5 inches. An 18-inch high curb will separate the flooded portion of the auxiliary building from the shield building where systems and components required for safe shutdown are located. The applicant is committed to design this curb to contain 2.5 inches of water and thus prevent its release to areas in which safety-related systems and components are located. Based on system flow rates and piping size and comparable isolation capability, we agree with the applicant that leakage cracks in other piping that penetrate the wall drain system should be less severe than for the nuclear service water pipe break analysis assuming comparable isolation times.

As a result of our review, we conclude that the design criteria and bases for the permanent dewatering system meet the single failure criterion for safety-related systems. On the bases of the applicant's commitment to limit flooding to a 2.5 inch height within a limited area of the auxiliary building, and that the design will include a barrier to isolate this amount of water from safety-related systems and components, we find the system acceptable. Consideration of other sources of flooding is discussed in Section 2.4.5 of this report.

712 140

10.0 STEAM AND POWER CONVERSION SYSTEMS

10.1 Summary Description

The scope of the standard reference nuclear steam supply system design described in the CESSAR includes the steam generator. The remainder of the steam and power conversion system design is within the balance of plant scope. The interface areas between the steam generator and the balance of plant design are at the steam generator feedwater and steam nozzles.

The steam and power conversion system will be of conventional design similar to those of previously approved plants. The system will be designed to remove thermal energy from the reactor coolant by two steam generators and convert it to electrical energy by the turbine-driven generator. The condenser will transfer unusable heat in the cycle to the circulating water. The entire system will be designed for the maximum expected thermal output from the nuclear steam supply system.

In the event of a turbine trip or a large load reduction, the heat to the steam generators will be dissipated via the turbine bypass system to the condenser or through relief valves to the atmosphere if the condenser is not available.

10.2 Turbine Generator

As discussed in Section 3.5.3 of this report, the three turbine generators will be arranged in a peninsular orientation relative to their respective reactor buildings, to protect safety-related systems against damage from turbine missiles.

The turbine electro-hydraulic control system will control the speed of the turbine (1800 revolutions per minute, rated) by modulating the turbine inlet steam control valves to regulate the steam flow of the turbine.

The turbine control system will be designed to trip the turbine under the following conditions: turbine overspeed, loss of condenser vacuum, excessive thrust bearing wear, reactor trip, generator electrical trip, high exhaust hood temperature, low bearing oil pressure, and manual trip from the control room or at the turbine.

The turbine generator will be provided with two independent overspeed protection systems. These devices will trip the turbine at 110 and 111.5 percent of turbine rated speed, respectively, by closing the turbine stop, control, reheat stop and interceptor valves. Because of the redundancy in the turbine overspeed protection system, the turbine is, therefore, protected from excessive overspeed.

We have reviewed the adequacy of the applicant's proposed design criteria and design bases for operation of the turbine generator under normal, abnormal and accident conditions. Based on our review, we have concluded that the design criteria and design bases are acceptable.

10.3 Main Steam Supply System

The steam generated in each steam generator will be routed through two steam lines to the turbine. Each of these four main steam lines will contain four ASME Code, spring-loaded safety valves, one air-operated relief valve, and one main steam isolation valve. The main feed pump turbine steam supply will be located on the downstream side of the main steam isolation valves. Steam for each of the two turbine-driven auxiliary feedwater pumps will be taken from the steam generator via one of the two main steam lines upstream of the isolation valves. The steam piping and associated supports from the steam generators to and including the isolation valves will be seismic Category I. The remaining steam piping will be in accordance with ANSI B31.1. The valves and piping within the structure will be restrained to prevent pipe whip damage if a pipe break occurs. A failure of any main steam line or malfunction of a valve in the system will not prevent any of the engineered safety feature equipment from performing its function.

The main steam isolation valves will be located outside containment in a missile protected enclosure and will have an operator stroke time of five seconds or less after receiving a main steam line actuation signal. The valve will be designed to close for the condition of the maximum mass flow rate in either direction in the event of a double-ended steam line break. Failure of one main steam isolation valve coincident with a steam line break will not result in uncontrolled steam blowdown from more than one steam generator, based on proposed design main steam isolation valves leakage rates.

The plant capability to achieve safe cold shutdown in the event of a main steam line break with simultaneous loss of offsite power will be protected since each redundant set of main steam isolation valves, safety valves, atmospheric dump valves, and the supply piping up to and including the turbine-driven auxiliary feed pump will be housed in a separate auxiliary building enclosure designed to withstand the failure of high energy lines. The relief valves connected to the unaffected steam lines will be manually operated to decrease primary and secondary plant pressure at a rate that is compatible with initiation of the residual heat removal system which is then utilized to remove the decay heat.

As a result of our review, we conclude that the main steam supply system design criteria and bases are in conformance with the single failure criterion, the seismic design position of Regulatory Guide 1.29, "Seismic Design Classification," and valve closure time requirements, and are, therefore, acceptable.

10.4 Circulating Water System

The circulating water system will be designed to use water from the cooling towers to remove heat rejected from the main condensers and feedwater pump turbine condensers.

A failure in the circulating water system or the condensate system large enough to cause flooding will be detected by high level alarms in the turbine room sumps and condenser pits. The alarm will alert the operator to take action in isolating the equipment or shut down the system completely. There will be no safety-related equipment in the turbine building that can be affected by flooding. Measures to protect other portions of the plant against the effects of flooding in the turbine building are discussed in Section 2.4.5 of this report.

Based on our review of the design criteria and design bases for the circulating water system, we conclude that this system will perform its intended function and therefore is acceptable.

10.5 Feedwater Systems

10.5.2 Auxiliary Feedwater System

The auxiliary feedwater system will be designed to independently supply water to the steam generators to remove sensible and decay heat when the main condensate and feedwater system is not available. The auxiliary feedwater system will be designed to function automatically during certain periods of normal startup and shutdown, in the event of malfunctions such as loss of onsite and offsite power, loss of normal feedwater and in the event of accidents such as a steam line rupture. The system will be designed to seismic Category I requirements and will be protected from tornado missiles.

The auxiliary feedwater system will consist of two independent trains. Each train will include one 100 percent capacity (875 gallons per minute) motor-driven pump, one 100 percent capacity steam-driven pump, one feedwater supply tank, associated valves, piping and instrumentation.

The motor-driven pumps will be powered from the emergency diesel generator buses. Those valves, equipped with electric operators, that are required to function during emergency operation will also be powered from these buses. Steam to each drive turbine will be supplied from a main steam line upstream of the main steam isolation valve. The flow control valves in these lines and the valves on the discharge side of the pump will be pneumatically operated and actuated by direct current power. Both valves will function automatically upon receipt of an auxiliary feedwater actuation signal. Valves equipped with pneumatic operators will be provided with accumulators to assure 30 minutes of operational time. Handwheels will be provided for backup.

The auxiliary feedwater system as described will be designed in accordance with the power diversity provisions of Auxiliary and Power Conversion Systems Branch Technical

Position APCSB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants."

The auxiliary feedwater system will be designed to accomplish its safety function in the event of a high energy line failure coincident with a single active failure. Crossover lines with two valves that fail closed will be provided between trains to assure isolation of the affected steam generator in the event of a steam line rupture inside containment. Within each train, the motor-driven and steam-driven pump will be physically separated.

The applicant performed an analysis that demonstrates that adequate decay heat removal will be obtained with a minimum of one pump and one steam generator. We agree with the conclusions of that evaluation. The design of the system assures that at least one pump and one steam generator will be available assuming the combination of a single active and a high energy line failure during all operating conditions.

The auxiliary feedwater system will be supplied from two auxiliary feedwater storage tanks located in the auxiliary building if water is not available from the non-seismic Category I condensate storage tank. Each auxiliary feedwater pump will be supplied through an individual supply line from its storage tank. Each auxiliary feedwater tank will contain a minimum of 40,000 gallons of condensate for a total of 80,000 gallons. This total volume can satisfy the steam generator feedwater requirement for approximately one hour with both auxiliary feedwater system trains in operation or for approximately 30 minutes with one train's tank out of service in the event of a single failure. However, prior to depleting the condensate supply in the auxiliary feedwater tanks, operator action can be used to actuate two motor-operated valves and admit water to the auxiliary feedwater pumps from the nuclear service water system. This water is available in sufficient quantity to maintain the plant at hot shutdown for two hours, followed by cooldown to a condition at which the residual heat removal system can be initiated.

The Class IE alternating current and direct current power provided for one train will be separate and independent of the Class IE supply for the redundant train. All breaker control power will be direct current.

As a result of our review of the proposed design, we conclude that these design criteria will satisfy our requirements for the design of auxiliary feedwater systems and the criteria discussed in Section 8.1 of this report, and are, therefore, acceptable.

The applicant has modified his original design to provide two supply valves in parallel from each auxiliary feedwater tank. We have concluded that the design now satisfies the criteria for instrumentation and control outlined in Section 7.1.1 of this report and is, therefore, acceptable.

Damage to the auxiliary feedwater system piping such as occurred at Indian Point 2 on November 13, 1973, could originate as a consequence of uncovering of the feedwater sparger in the steam generator or uncovering the steam generator feedwater inlet nozzles. Uncovering the steam generator feedwater nozzles could cause a pressure wave that is propagated through the piping. We are conducting a generic review of this problem. We will consider the results of our generic review of feedwater flow instability in determining a final position on the matter for the proposed facility during the operating license stage of our review.

We reviewed the adequacy of the applicant's proposed criteria and design bases for the auxiliary feedwater system to assure safe operation of the plant during normal, abnormal, and accident conditions. Based on our review, we have concluded that the design criteria and design bases for this system conform to our technical positions regarding diversity of power sources, system flexibility and redundancy including the combination of single active and high energy line failures and are, therefore, acceptable.

10.6 Steam and Feedwater System Materials

The mechanical properties of materials selected for the Class 2 and 3 components of the steam and feedwater systems will satisfy Appendix I of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and Parts A, B and C of Section II of the ASME Boiler and Pressure Vessel Code.

The controls imposed upon the austenitic stainless steel generally satisfy the recommendations of Regulatory Guide 1.31, "Control of Stainless Steel Welding," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices that will be performed in accordance with these recommendations provide added assurance that stress-corrosion cracking will not occur during the design life of the plant. The controls that will be placed upon concentrations of leachable impurities in nonmetallic thermal insulation used on austenitic stainless steel components of the steam and feedwater systems are in accordance with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

The welding procedures that will be used in limited access areas generally satisfy the intent of the recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility." The onsite cleaning and cleanliness controls during fabrication satisfy the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." The precautions taken in controlling and monitoring the preheat and interpass temperatures during welding of carbon and low alloy steel components conform to the recommendations given in Regulatory Guide 1.50, "Control of Preheat Temperatures for Welding of Low-Alloy Steel."

In Amendment 28, the applicant identified some exceptions to the recommendations of Regulatory Guides 1.31, 1.44 and 1.71 that he proposes to adopt. We will report the results of our review of this matter in a supplement to this report, and if significant exceptions remain, we will require that the applicant either conform with the recommendations of the guides or provide alternate acceptable bases for meeting our licensing requirement prior to a decision on issuance of construction permits.

Conformance with the [illegible] standards, and applicable Regulatory Guides cited above, if provided by the applicant, constitutes an acceptable basis for assuring the integrity of steam and feedwater systems, and for meeting the requirements of Criterion 1 of the General Design Criteria.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

Each unit will have its own, completely independent, radioactive waste management system designed to provide for controlled handling and treatment of liquid, gaseous and solid wastes. Each liquid waste system will process wastes from equipment and floor drains, decontamination and laboratory wastes and laundry and shower wastes. Each gaseous waste system will provide holdup capacity to decay short-lived noble gases stripped from the primary coolant and treatment of ventilation exhausts through high efficiency particulate air filters and charcoal adsorbers. The systems will be designed to reduce releases of radioactive materials in effluents to "as low as is reasonably achievable" levels in accordance with Section 50.34 of 10 CFR Part 50. Each solid waste system will provide for the solidification, packaging and storage of radioactive wastes generated during facility operation prior to shipment offsite for burial. Solid packaged wastes will be shipped to a licensed facility for burial.

In our evaluation of the liquid and gaseous radwaste systems, we have considered: (1) the capability of the systems for keeping the levels of radioactivity in effluents "as low as is reasonably achievable," based on expected radwaste inputs over the life of the plant, (2) the capability of the systems to maintain releases below the limits in 10 CFR Part 20 during periods of fission product leakage at design levels from the fuel, (3) the capability of the systems to meet the processing demands of the station during anticipated operational occurrences, (4) the quality group and seismic design classification applied to the system design, (5) the design features that will be incorporated to control the releases of radioactive materials in accordance with Criterion 60 of the General Design Criteria, and (6) the potential for gaseous release due to hydrogen explosions in the gaseous radwaste system.

In our evaluation of the solid radwaste treatment systems we have considered: (1) system design objectives in terms of expected types, volumes and activities of waste processed for offsite shipment, (2) waste packaging and conformance to applicable federal packaging regulations, and provisions for controlling potentially radioactive airborne dusts during baling operations, and (3) provisions for onsite storage prior to shipping.

In our evaluation of the process and effluent radiological monitoring and sampling systems, we have considered the system's capability: (1) to monitor all normal and potential pathways for release of radioactive materials to the environment, (2) to control the release of radioactive materials to the environment, and (3) to monitor the performance of process equipment and detect radioactive leakage between systems.

Since the Final Environmental Statement for the facility was issued in October 1975, the applicant has modified the liquid, gaseous and solid radioactive waste management systems by amendments to the PSAR. A summary of these modifications is shown in Table 11.1 of this report. In addition, we stated in the Final Environmental Statement that to effectively implement the requirements of Appendix I to 10 CFR Part 50, we would reassess the parameters and mathematical models used in calculating releases of radioactive materials in liquid and gaseous effluents considering current operating data in the assessment of the input parameters. The Atomic Safety and Licensing Board issued its Order on January 13, 1977, modifying its partial Initial Decision as to environmental and site suitability issues by deleting paragraph 115 (IV) (NRC-76/5 page 651) and substituting the following in lieu thereof: "Duke Power Company shall not remove any major components of the radwaste treatment system without replacing them with components to maintain equivalent overall system performance capability." We have completed our review of changes in the waste management systems proposed by the applicant in Amendments 22 and 23 to the PSAR and conclude that with these changes the overall system performance capability will be equivalant to the capability of the system as proposed in the PSAR as amended to August 8, 1975 (the date after which the Board's Order of January 13, 1977 applies).

The parameters, models and their bases that resulted from our reassessment are provided in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors" (PWR-GALE Code), April 1976. Based on information provided by the applicant in Amendment 23 to the PSAR, and considering the modifications in Table 11.1, we have recalculated the quantities of radioactive materials that will be released in liquid and gaseous effluents and the quantity of material that will be shipped as solid radwaste for burial during normal operation of the facility. In making these determinations, we considered waste flows, activities and equipment performance consistent with normal plant operation, including anticipated operational occurrences, over the life of the plant. Liquid and gaseous source terms were recalculated using the PWR-GALE Code. These new source terms and input parameters used for the facility were incorporated in testimony presented at the environmental hearings. This testimony, entitled "NRC Staff Evaluation of Liquid and Gaseous Effluents with Respect to Appendix I of 10 CFR Part 50" for the facility is provided in Appendix E to this Safety Evaluation Report. Based on our evaluations, the proposed liquid and gaseous radwaste treatment systems, which are the same as those evaluated in the testimony for the facility, meet the criteria given in Appendix I to 10 CFR Part 50 and are, therefore, acceptable.

11.2 Liquid Radwaste Treatment System

Each liquid radioactive waste treatment system will consist of process equipment and instrumentation necessary to collect, process, monitor and recycle or dispose of liquid radioactive wastes. The liquid radioactive waste will be processed on a batch basis to permit optimum control of releases. Prior to being released, samples

TABLE 11.1

MODIFICATIONS TO THE WASTE MANAGEMENT SYSTEMS
AS PROPOSED IN THE PSAR AMENDMENTS
SINCE THE FINAL ENVIRONMENTAL STATEMENT (FES)

	Description in the FES	Modification Description	PSAR Amendment No.
1.	Two Laundry and Hot Shower Tanks, 4000 gallons per tank	Two Laundry Tanks, 8000 gallons per tank	11 and 13
2.	Two Waste Monitor Tanks, 15,000 gallons per tank	Four Waste Condensate Tanks, 27,700 gallons per tank	13
3.	Two Waste Holdup Tanks, 15,000 gallons per tank	Four Waste Tanks, 27,700 gallons per tank	13 and 14
4.	One Waste Evaporator Package, 20 gallons per minute	Two Waste Concentrators operating in parallel, 25 gallons per concentrator	13
5.	One Volatile Chemistry Control (VCC) System on the secondary coolant loops (Figure 3.7 in FES)	Five condensate polishing filter/demineralizers for volatile chemistry control, approximately 4450 gallons per minute for each filter/demineralizer (four normally in operation with one in backwash). Backwash separator tank with provisions for sampling, monitoring and control of potentially radioactive wastes.	11 and 14
6.	One Concentrate Hold Tank, 1000 gallons	Two Waste Concentrate Tanks, 5000 gallons per tank	11 and 13
7.	One Spent Resin Storage Tank, 5000 gallons	Two Spent Resin Storage Tanks, 5000 gallons per tank	11
8.	Gas Collection Header (Figure 3.8 in FES)	Provisions for treating the vent wastes from the Holdup Tank, Waste Tanks, Equipment Drain Tank, Refueling Water Tank and the Concentrate Tanks by the Auxiliary Building filter/charcoal train prior to release.	22
9.	Release Points	Additional information in accordance with the option 1975 Amendment to Section II.D of Appendix I	3, 14 and 23

712 149

will be analyzed to determine the types and amounts of radioactivity present. Based on the results of the analysis, the waste will be retained for further processing, recycled for eventual use in the plant, or released under controlled conditions. Aerated radioactive wastes will be segregated, based on their origin, and processed through the liquid waste processing system. Turbine building floor drain wastes will be discharged without treatment unless sampling indicates processing through the liquid waste processing system is necessary. Detergent (laundry and decontamination) wastes will be sampled, filtered and discharged or processed by the liquid waste management system when sampling indicates processing is necessary. The principal components making up each of these systems, along with their principal design parameters, are listed in Table 11.2.

The design capacity of each of the two liquid waste processing system evaporators will be 36,000 gallons per day. We calculated that the average expected waste flow to the liquid waste processing system will be 2100 gallons per day. The difference between the expected flow and design capacity will provide adequate reserve for processing surge flows. We consider the system capacity and system design to be adequate for meeting the demands of the station during anticipated operational occurrences.

Blowdown from the steam generators will be recycled to the main condenser through condensate polishing filter/demineralizers in the secondary system. Normally, the filter/demineralizers will be backwashed once every two to five days to a polishing demineralizer backwash holding tank. Backwash waste will be continuously monitored for radioactivity, liquids will be transferred to the liquid waste treatment system and backwash sludge will be transferred to the solids treatment system if the activity exceeds a predetermined value. The applicant will be required by technical specifications to take batch samples and analyze the solid and liquid wastes for potential activity prior to controlled release to the waste water holdup basin or transfer to the radwaste treatment systems. There will be no steam generator blowdown waste release.

The liquid radwaste systems will be located in a seismic Category I structure. The liquid radwaste system components, capacities and seismic and quality group classifications proposed by the applicant are listed in Table 11.2. The system will also be designed to control the release of radioactive materials due to overflows from indoor and outdoor tanks by providing level instrumentation which will alarm in the control room, and by means of curbs and retention walls to collect liquid spillage and retain it for processing. We consider these provisions to be capable of preventing the uncontrolled release of radioactive materials to the environment. We find the applicant's proposed system design to be in accordance with the staff's technical position as shown in Branch Technical Position ETSB 11-1, "Design Guidance for Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Reactor Plants." Therefore, we conclude that the proposed design is acceptable.

TABLE 11.2

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS CONSIDERED IN THE RADWASTE EVALUATION

Components[1/]	Number	Capacity Each
Liquid Waste Management System		
Waste Concentrators	2	25 gallons per minute
Containment Cooler Condensate Tanks	2	4,000 gallons
Laundry Tanks	2	8,000 gallons
Waste Tanks	4	27,700 gallons
Waste Condensate Ion Exchange	2	20 gallons per minute
Waste Condensate Tanks	4	30 cubic feet
Gaseous Waste Management System		
Compressors	2	2 standard cubic feet per minute
Surge Tank	1	20 cubic feet
Decay Tanks	3	700 cubic feet
Recombiner	1	2 standard cubic feet per minute
Solid Waste Management System		
Spent Resin Tanks	2	5,000 gallons
Waste Concentrate Tanks	2	5,000 gallons

[1/] Design code and seismic design criteria in accordance with staff position in Branch Technical Position ETSB 11-1.

We have determined that, during normal operation, the proposed liquid radwaste treatment systems will be capable of reducing the release of radioactive materials in liquid effluents to approximately 0.19 curies per year per reactor, excluding tritium and dissolved gases, and 750 curies per year per reactor for tritium.

Based on our evaluation we find the proposed liquid radwaste system to be acceptable.

11.3 Gaseous Radwaste Treatment System

Each gaseous radwaste treatment system will be designed to process gaseous plant wastes based on the origin of the wastes in the plant and the expected activity levels. The gaseous waste treatment system will consist of a gaseous waste management system, a main condenser effluent processing system, and ventilation systems that control the release of radioactive effluents to the environment. The principal components of the system, along with their principal design parameters, are listed in Table 11.2.

The gaseous waste management system will collect and process gases stripped from the primary coolant along with miscellaneous tank cover gases contained with nitrogen in a loop provided for continuous recirculation. Operating with one of the two, two standard cubic feet per minute compressors and three 700-cubic foot gas decay tanks (each of which is capable of being isolated from all others), the gaseous waste management system will have adequate capacity to allow operation during periods of equipment downtime. We consider the system capacity and the system design to be adequate for meeting the demands of the station during normal operations and anticipated operational occurrences. The system design criteria and locating the gaseous waste treatment system in a seismic Category I structure are in accordance to the staff position, shown in Branch Technical Position ETSB 11-1. We find the system quality group and seismic design classification to be acceptable.

The system will be designed to operate at positive pressure and will be purged with nitrogen gas to prevent air (oxygen) buildup as a result of infiltration. Hydrogen and oxygen concentrations in gases entering the gaseous waste management system and stored in the decay tanks will be monitored by an automatically sequenced gas analyzer. The quantity of hydrogen or oxygen present will be reduced by passing the stored gas through a catalytic recombiner to form water. The gas analyzer will indicate concentrations of hydrogen and oxygen in sufficient time to allow purging the system with nitrogen gas before potentially explosive mixtures could occur. We found that the use of a single gas analyzer is not acceptable. The applicant by a letter dated February 8, 1977, has committed to provide dual gas analyzers with automatic control functions to monitor the formation or buildup of potentially explosive mixtures of hydrogen and oxygen. The two analyzers will operate continuously to provide two independent measurements and will alarm both locally and in the control room. On the basis that we find this information acceptable and that the applicant has committed to include the same information in Amendment 29 to the PSAR, we conclude that an acceptable local gas analyzer system will be provided in the facility design.

Gaseous wastes from the main condenser will be processed through filters and charcoal adsorbers for particulate and iodine removal, respectively. Noble gases will not be affected by the treatment provided.

Ventilation exhausts from the containment building and the auxiliary building, including the radwaste and fuel handling areas, will be processed through high efficiency particulate air filters and charcoal adsorbers prior to release. In addition, the containment building atmosphere will be recirculated through filters and charcoal adsorbers prior to purging to the ventilation exhaust system. The turbine building ventilation exhausts will be released to the environment without treatment. The plant ventilation systems will be designed to induce air flows from potentially less radioactive contaminated areas to areas having a greater potential for radioactive contamination.

We have determined that the proposed gaseous radwaste treatment systems and plant ventilation systems will be capable of reducing the release of radioactive materials in gaseous effluents to approximately 6700 curies per year per reactor for noble gases, 0.008 curies per year per reactor for iodine-131, 760 curies per year per reactor of tritium, eight curies per year per reactor for carbon-14 and 0.043 curies per year per reactor for particulates.

Based on our evaluation, we find the proposed gaseous radwaste system to be acceptable.

11.4 Solid Radwaste Treatment System

Each solid radwaste treatment system will be designed to collect and process wastes based on their physical form and need for solidification prior to packaging. "Wet" solid wastes, consisting of spent demineralizer resins, evaporator bottoms, filter sludges, and chemical drain tank effluents, will be combined with a solidification agent and catalyst to form a solid matrix and sealed in the shipping containers. Dry solid wastes, consisting of ventilation air filters, contaminated clothing and paper, and miscellaneous items such as tools and glassware, will be compacted into 55-gallon steel drums. Miscellaneous solid wastes, such as irradiated primary system components, will be handled on a case-by-case basis based on their size and activity. Expected solid waste volumes and activities shipped offsite for each reactor will be 3200 drums per year of "wet" solid waste containing an average of 0.6 curies per drum and 600 drums per year of "dry" solid waste containing less than five curies total.

712 153

Drum filling operations will be controlled remotely from consoles located outside the drum fill area. Drumming operations will have interlock features to prevent opening of filling valves when a drum is not properly positioned in the filling station. Baling of dry wastes will be carried out inside a closed and vented dust shroud. The shroud will be vented through high efficiency particulate air filters to the unit vent.

The solid radwaste systems will be located in a seismic Category I structure. The seismic and quality group designations of the equipment are consistent with our guidelines. The design parameters for the solid waste system component are listed in Table 11.2.

Storage facilities for up to 800 drums of solid radioactive wastes will be provided at plant grade in the radwaste building. Based on our estimate of 3800 drums per year per reactor, we find the storage capacity adequate for meeting the demands of the station. Wastes will be packaged in 55-gallon steel drums in accordance with the requirements of 10 CFR Part 20 and 10 CFR Part 71, and shipped to a licensed burial site in accordance with Commission and Department of Transportation regulations.

Based on our evaluation, we find the proposed solid radwaste treatment system to be acceptable.

11.5 Process and Effluent Radiological Monitoring Systems

The process and effluent radiological monitoring and sampling systems will be designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in plant discharges to the environs. Liquid and gaseous streams will be monitored. Table 11.3 indicates the proposed locations and types of continuous monitors. Monitors on certain effluent release lines will automatically terminate discharges should radiation levels exceed a predetermined value. Systems, which are not amenable to continuous monitoring or for which detailed isotopic analyses are required, will be sampled and analyzed in the plant laboratory. We have reviewed the locations and types of effluent and process monitoring provided.

Based on the plant design and on the continuous monitoring locations and continuous and intermittent sampling locations, we have concluded that all normal and potential release pathways will be monitored. We have also determined that the sampling and monitoring provisions will be adequate for detecting radioactive material leakage to normally uncontaminated systems and for monitoring plant processes which affect radioactivity releases. On this basis, we consider that the monitoring and sampling provisions meet the requirements of Criteria 13, 60 and 64 of the General Design Criteria and the guidelines of Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in

TABLE 11.3

PROCESS AND EFFLUENT MONITORING

Stream Monitored

Liquid[1]

Component Cooling Water

Reactor Coolant

Liquid Waste Releases (Plant Effluents)[2]

Gas[1]

Containment Purge and Vent[2]

Unit Vent

Condenser Air Ejector Exhaust

Radwaste Area Exhaust

Waste Gas Discharge[2]

Auxiliary Building

Spent Fuel Building

---

[1] All liquid and gas streams will be monitored in accordance with the guidelines of Regulatory Guide 1.21.

[2] These monitors provide annunciation and automatic closure of isolation valves terminating releases when the radiation level exceeds a predetermined value.

712 155

Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," and are, therefore, acceptable.

11.6 Evaluation Findings

Our review of the radwaste systems included a review of system capabilities to process the types and volumes of wastes expected during normal operations and anticipated operational occurrences in accordance with Criterion 60 of the General Design Criteria, the design provisions incorporated in accordance with Criterion 60 to control releases of radioactive material due to leakage overflows, the quality group and seismic design classification in conformance with the guidelines of Branch Technical Position ETSB 11-1, "Design Guidance for Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants," and the design provisions incorporated in conformance with the guidelines of Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable (Nuclear Power Reactors)," paragraph C.3. We have reviewed the applicant's system descriptions, process flow diagrams, piping and instrumentation diagrams, and design criteria for the components of the radwaste treatment systems and for those auxiliary supporting systems that are essential to the operation of the radwaste treatment systems. We have performed an independent calculation of the releases of radioactive materials in liquid and gaseous effluents based on calculational methods contained in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," April 1976.

Our review of the process and effluent radiological monitoring and sampling systems included the provisions proposed for sampling and monitoring all station effluents in accordance with Criterion 64 of the General Design Criteria for providing automatic termination of effluent releases and assuring control over discharges in accordance with Criterion 60 and Regulatory Guide 1.21, for sampling and monitoring plant waste process streams for process control in accordance with Criterion 63 of the General Design Criteria, for conducting sampling and analytical programs in accordance with the guidelines in Regulatory Guide 1.21, and for monitoring process and effluent streams during postulated accidents. The review included piping and instrument diagrams and process flow diagrams for the liquid, gaseous, and solid radwaste systems and ventilation systems, and the location of monitoring points relative to effluent release points on the site plot diagram.

Based on the foregoing evaluation, we conclude that the above aspects of the proposed radwaste treatment and monitoring systems are acceptable. The basis for acceptance has been conformance of the applicant's designs, design criteria, and design bases for the radioactive waste treatment and monitoring system to the applicable regulations and guides referenced above, as well as to staff technical positions and industry standards.

The capability of the liquid and gaseous radioactive waste treatment systems to meet the dose design objectives of Appendix I to 10 CFR Part 50 and the required cost-benefit analysis were evaluated for testimony in the hearing using the same terms, input parameters, and models that we have reviewed and found acceptable as described above. That testimony is reproduced as Appendix E to this report.

12.0 RADIATION PROTECTION

The applicant has provided descriptions of methods for radiation protection and has included an estimate of occupational radiation doses to plant personnel. The PSAR presents information on facility layout, equipment design, operating procedures, techniques, and practices proposed for the protection of personnel against radiation. Shielding will be provided to reduce levels of radiation. Ventilation will be arranged to control the flow of potentially contaminated air. Radiation monitoring systems will be employed to measure levels of radiation in potentially occupied areas and to measure airborne radioactivity throughout the plant. A health physics program will be provided for plant personnel and visitors during reactor operation, maintenance, refueling, radwaste handling and inservice inspection.

We reviewed and evaluated the applicant's description and analysis of the radiation protection program, contained in Section 12.0 of the PSAR. The criteria used to determine acceptability of the applicant's program are that doses to personnel will be maintained less than those limits established in 10 CFR Part 20, and that design and program features are consistent with the guidelines of Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable."

On the basis of our review, we have concluded that the radiation protection program will provide assurance that doses to personnel will be less than those limits established by 10 CFR Part 20 and that design features and program features are consistent with the guidelines of Regulatory Guide 8.8. The applicant's overall preliminary radiation protection program is acceptable. The details of our review are discussed in the following sections.

12.1 Shielding

The design objectives for the facility shielding are to ensure that radiation exposure to operating personnel will be within the required limits of 10 CFR Part 20 and that these exposures will also be maintained as low as reasonably achievable during reactor operations and surveillance, maintenance, inservice inspections, refuelings and radwaste handling.

Plant areas have been classified into radiation zones based on expected frequency and duration of occupancy. The design of the radiation shielding will consider the dose rate criterion for each zone based on maximum short-term radiation sources in each compartment within the zone. All radioactive sources that form the bases for the shield design have been considered. Shielding analysis will be made using accepted codes, models and assumptions. A check-off list which contains design guidelines, as given in Regulatory Guide 8.8, and relevant facility and equipment deficiencies from other reactors, will be used in making design reviews to assure that the shielding will be designed to permit limiting radiation exposures to levels that are as low as reasonably achievable.

Consistent with the design, the applicant has addressed the steps that will be taken to assure that low dose rate zones will not be compromised by inadvertent increases in radiation levels. Consequently, pipes carrying radioactive liquids including field run piping, filters, demineralizers, tanks, evaporators, pumps and sampling points will be designed to be located in shielded compartments. Tanks within compartments that can contain significant quantities of radioactivity will be shielded from each other. Therefore, each component or tank within a compartment will be isolated to allow maintenance, inspection, and some non-routine operations with radiation interference from other components or tanks that is as low as reasonably achievable. Labyrinths will be used for entranceways to cubicles to retain shielding integrity. In addition, shielded valve galleries, shielded penetrations, reach rods, remote switching and portable shielding among other devices, will be used to maintain exposures to levels that are as low as reasonably achievable.

On the basis of the applicant's design criteria, shield models and operating philosophy, we conclude that adequate consideration has been given in the PSAR to the shielding and layout of facilities and components to keep exposures to operating personnel within the applicable limits of 10 CFR Part 20, and to reduce unnecessary exposure during normal operation of the facility, including the considerations stated in Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as is Reasonably Achievable."

12.1.1 Area Monitoring

The radiological monitoring system will be designed to continuously measure the radiation levels at 14 selected locations within the plant. Each will be a location having a potential for both significant radiation levels and occupancy. Each instrument of the system will have a sensor and an audible alarm at the fixed location where personnel perform work on a regular basis, and audible and visual annuciation in the control room. Radiation levels will be recorded on a multi-point recorder in the control room. Each detector will be equipped with a check-point source and controls necessary to operate it from the control room to verify the response of detector read-out and alarm channels.

12.2 Ventilation

The ventilation systems will be designed to provide a suitable radiological environment for personnel and equipment, and to assure compliance with the limits for restricted areas set forth in 10 CFR Part 20. Air flow will be from areas of low radioactivity toward areas of higher activity to prevent the spread of airborne radioactive material and thereby ensure contamination control. Ventilation design considerations for atmosphere clean up systems are described in Section 9.4 of this report.

Various compartments throughout the plant will be provided with roughing and high efficency particulate air filter banks, with charcoal filters added at selected locations, to preclude a buildup of airborne contamination. Provision will be made for special temporary local exhaust ventilation as required.

We conclude that the ventilation system will be based on design criteria that provide reasonable assurance that the system will be designed with the capability to maintain concentrations of airborne activity in areas normally occupied in accordance with 10 CFR Part 20.

12.2.1 Airborne Radioactivity Monitoring

Equipment for monitoring inplant airborne radioactivity will include: (1) fixed gas and particulate monitors located in the containment, the auxiliary building, the spent fuel building and radwaste area; (2) fixed iodine monitors located in areas where there is a potential for iodine-131 airborne activity; and (3) noble gas monitors located in the condensor air ejector and in the equipment and cable room ventilation system. Each of these monitors will include a pumping system for collecting samples. Particulates will be collected on a moving filter tape and counted with a plastic beta scintillator. Noble gases and iodines will be monitored with sodium iodide detectors. These detectors will be capable of detecting fractions of maximum permissible concentrations. Output information is displayed and recorded in the control room and if the output exceeds a selected level an alarm is initiated. Alarms will also be initiated by loss of air flow to the monitors. The containment airborne monitor will draw samples through a manifold from various locations in containment, including upper and lower containment regions, incore instrument room and containment purge. The source will be controlled from the control room by solenoid-operated sample valves. The auxiliary building monitoring system also will sample from 12 individual locations within the building through the use of sequential solenoid valves. This scanning system thus provides coverage in many areas with a small number of pumping systems and detectors. Location of the sample points will be described in the FSAR.

We conclude that the scope of the area monitoring program will provide satisfactory information for use in providing radiological protection to in-plant personnel. Area radiation detectors are to be located in areas that have a potential for radiation fields in excess of radiation zone designations. Airborne radioactivity monitors will be located in ventilation ducts and compartments where there is a likelihood for inadvertent release of airborne radioactivity. State-of-the-art sensitivity and alarm annuciation techniques will be used in the design of the monitoring systems.

The applicant's proposed use of a multi-point sampling system for the auxiliary building is acceptable and should provide a satisfactory airborne radioactivity system for that building. At the operating license stage of our review, we will review the sample location points and sampling design to ascertain that airborne concentration measurement errors, associated with excessive and non-uniform particulate deposition in the sample lines, are not being made.

12.3 Health Physics

Our review of the applicant's health physics program covered management policies, organizational structure and program for maintaining occupational exposures as low as reasonably achievable. We reviewed the health physics program, facilities and monitoring equipment, and procedures related to contamination control and occupational radiation exposures. The applicant's stated policy for radiation protection is based on compliance with appropriate regulations, use of applicable Regulatory Guides and development of appropriate technical specifications. The radiation protection equipment will include personnel thermoluminescent dosimeters and/or film badges protective clothing and respiratory equipment. Radiation exposure control measures will include barriers, locked doors, signs, audible and visible indicators and alarms, and other access control measures to preclude unauthorized entry into radiation control areas, use of special work permits and procedures, testing and calibration of monitoring instrumentation, and maintenance of radiological reports and records.

The radiation protection facilities will include a shielded counting room for counting air and swipe samples, an instrumentation calibration room for checking health physics survey instruments, a change room for clean protective clothing and respirators, and a personnel and equipment decontamination room. The counting room will contain a multi-channel pulse height analyzer with associated sodium iodide and germanium lithium detectors, beta-gamma counter-scalers, scintillation systems for counting alpha and tritium, and a shielded body-burden thyroid-burden analyzer used for bioassay purposes. A thermoluminescent reader and associated equipment will be provided in the counting room for use in radiation surveys and personnel dosimetry.

Health physics personnel will review and maintain a continuing evaluation of radiation levels in all areas where personnel will be working. Instruments to be used for radiation surveys consist of alpha, beta, gamma and neutron survey meters. Samplers for airborne gases, particulates and iodines, continuous air monitors, bubblers for tritium, gas-sample containers and low and high volume air samplers will be available. For contamination control, fixed and portable radiation instruments will be used as portal monitors at exits from radiation control areas and to monitor personnel leaving the station.

All personnel whose job involves radiation exposure as defined in 10 CFR Part 20, Section 20.202, will be provided with personnel monitoring equipment. Neutron sensitive film will be worn as required by plant conditions. Pocket chambers and dosimeters, pocket high-radiation alarms, wrist badges and finger tabs, will be available and will be used under specified radiation work permit conditions.

Pocket chambers and dosimeters will be maintained by the health physics staff for recording daily exposures. Dosimeter records will be used as a source of exposure data for use in administrative control of radiation exposure. Routine body-burden analysis will be performed on a portion of personnel who work in radiation areas and who have the highest exposure potential. The applicant estimates that inhalation doses will result in-plant personnel exposures of 0.8 man-rem per year to the whole body and 0.4 man-rem per year to the thyroid. Body burden scans will be given to anyone involved in a radiological accident.

On the basis of the plant design criteria, health physics related equipment and procedures, and the applicant's consideration of the recommendations of Regulatory Guide 8.8, we conclude that the applicant's health physics program will provide plant personnel with adequate protection against the radiation hazards associated with the normal operation of the plant and will limit occupational exposures to as low as reasonably achievable as required by 10 CFR Part 20.

13.0 CONDUCT OF OPERATIONS

13.1 Organization and Qualifications

The Duke Power Company is responsible for the design, construction and operation of the Cherokee Nuclear Station. Duke Power Company will act as its own architect engineer and be responsible for all site construction activities. Combustion Engineering, Incorporated, will design and manufacture the nuclear steam supply systems.

The Duke Power Company's Design Engineering Department will perform the architect engineering work and the Construction Department will direct the construction of the power generating facilities. The Vice President of each of these departments reports to the Senior Vice President, Engineering and Construction. The Steam Production Department will be responsible for the operation and maintenance of the Cherokee Nuclear Station. Quality assurance aspects of the project are discussed in Section 17.0 of this report.

The station organization for the operation of each of these facilities will consist of a technical staff of approximately 65 persons for one-unit operation, 107 persons for two-unit operation and 149 persons for three-unit operation under the direction of the Plant Manager. Reporting to the Plant Manager will be an Operating Superintendent who is responsible for directing the actual day-to-day operation of the station with a staff of up to 81 persons, a Technical Services Superintendent with a staff of up to 24 persons, and a Maintenance Superintendent who is responsible for directing plant maintenance activities with a staff of up to 41 persons. This is a conventional type of plant organization for providing onsite operating and technical support staff for plant operations. The shift crew for each unit of each station will consist of five persons, one of whom will hold a senior operator's license and two of whom will each hold an operator's license.

During most of our review, the applicant has indicated that the qualification requirements for the operating staff will be in accordance with American National Standards Institute N18.1 1971, "Selection and Training of Nuclear Power Plant Personnel." This would meet the staff's position stated in Regulatory Guide 1.8, "Personnel Selection and Training." In Amendment 25 the applicant revised his position to reduce the experience requirement for the Radiation Protection Manager from nine to seven years. We have concluded that the applicant's proposed dependence on support of the corporate health physicist and his staff does not provide a sufficient basis for the proposed reduction in experience requirement. Unless the applicant agrees to our position prior to the hearing, we will recommend to the Atomic Safety and Licensing Board that if construction permits are issued they be conditioned to include the staff position.

712 163

Technical support for the plant staff during plant operation will be provided by the Steam Production Department General Office staff. Other departments of the company will be available for assistance as necessary.

Except for the applicant's non-conformity with our position to require that nine years experience be a requirement for the Radiation Protection Manager, we conclude that the applicant has established an acceptable organization to design and construct the proposed facility and that his proposed plant organization, their proposed qualifications, and the plans for offsite technical support of plant operations are acceptable.

13.2 Training Program

The Vice President, Steam Production, has overall responsibility for the administration and conduct of the initial training program. At the station level the station manager is responsible for the training program and a station training coordinator directs the day-to-day administration and conduct of the program.

The applicant has stated that a comprehensive program will be conducted for the initial training of the station staff, with the objective of providing station personnel with the necessary skills and experience to startup, operate and maintain the station in a safe and efficient manner. The program to be used is similar to programs utilized at the applicant's Oconee and McGuire Nuclear Stations. Duke Power Company will conduct or contract for the teaching of each segment of the training program. Certain segments may be provided by North Carolina State University and/or Combustion Engineering Company.

The training provided for personnel to be licensed will include: selection examination, basic mathematics, nuclear preparatory, nuclear fundamentals, research reactor training, systems and procedures, observation training at an operating pressurized water reactor, reactor simulator training, and onsite training.

Maintenance and technical staff personnel will receive on-the-job training in specific skills. All station personnel receive general employee training consisting of training in station plans and procedures, radiological health and safety, industrial safety, controlled access areas and security procedures, and use of protective clothing and equipment.

Complete records of all training administered will be maintained.

On the basis of our review, we have concluded that the training program proposed for the facility will provide an acceptable number of trained personnel for operation of the facilities and is acceptable at the construction permit stage of review.

13.3 Emergency Planning

The applicant has described the preliminary plans for coping with emergencies. A more detailed emergency plan will be prepared and presented in the application for an operating license. The Shift Supervisor on duty will direct the implementation of the Emergency Plan in accordance with written emergency procedures.

For the Cherokee Nuclear Station, initial contacts and arrangements have been made with the South Carolina State Department of Health and Environmental Control, Division of Radiological Health; the Cherokee County Civil Preparedness Agency; the Sheriff's Department for Cherokee County, the Cherokee County Police; the South Carolina Highway Patrol; the Energy Research and Development Administration's Emergency Radiological Monitoring Team; the Nuclear Regulatory Commission's Region II Office of Inspection and Enforcement; and the Cherokee County Health Department. The South Carolina State Department of Health and Environmental Control will have primary responsibility for radiological emergency planning in the environs of the Cherokee site.

Communications equipment, instruments and controls for station operation will be provided in the control rooms. To aid in evaluation of any possible hazards offsite, the Shift Supervisor will utilize meteorological data available to the control rooms and information available from the station radiation monitoring system. He will also utilize meteorological overlays, nomographs or other calculational aids, local area maps and population data for this purpose. An emergency vehicle will be available for offsite monitoring. The control room in each unit will be designed for continuous occupancy. The control room in the affected unit will be the principal emergency control center. The Duke Power Company has designated its facility in Charlotte, North Carolinia, as an alternate emergency control center.

Decontamination facilities and a first aid room will be provided onsite. Preliminary contacts have been made with Gaston Memorial Hospital, Gastonia, North Carolina, and Memorial Hospital, Charlotte, North Carolina, to establish that agreements can be made and that potential capability exists for receiving and treating individuals that may be affected by radiological emergencies. Emergency transportation of individuals to the offsite treatment facilities will be provided by an emergency vehicle and outside ambulance and rescue services. All plant personnel will receive training in emergency procedures and periodic drills will be conducted. Offsite organizations will participate in the training programs.

The emergency and accident situations covered in each Emergency Plan include fires, vehicular accidents, natural disasters, medical injuries and illnesses, radiation and radioactive contamination incidents, and civil disturbances.

We have reviewed the applicant's preliminary plans for coping with emergencies and conclude that they meet the requirements of 10 CFR Part 50, Appendix E, and are acceptable for the construction permit stage of our review.

13.4 Review and Audit

The applicant has described his plans for the review and audit of the proposed plant operations. We have reviewed these plans and conclude that they generally meet those provisions described in American National Standards Institute N18.7-1972 "Administrative Controls for Nuclear Power Plants," and are acceptable for the construction permit stage of review.

13.5 Plant Procedures

All safety-related operating maintenance and testing activities will be performed in accordance with approved written procedures. American National Standards Institute N18.7-1972, "Administrative Controls for Nuclear Power Plants," and Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," will be used as guidelines in the development of applicable station procedures. Final approval of a procedure must come from the Station Superintendent.

All procedures except maintenance and periodic test procedures will be completed at least six months prior to fuel loading. Maintenance and periodic test procedures will be completed at a later date, but prior to fuel loading.

Based on our review, we conclude that the applicant's proposed program for preparation, review, approval and use of written procedures, and the commitment to document operating and maintenance activities are acceptable at the construction permit stage of review.

13.6 Plant Records

The applicant has described his plans for keeping plant records. We have reviewed these plans and conclude that they are generally in accord with those provisions described in American National Standards Institute N18.7-1972, "Administrative Controls for Nuclear Power Plants" and are acceptable for the construction permit stage of review.

13.7 Industrial Security

The applicant has provided a general description of plans for protecting the plant against potential acts of industrial sabotage. Provisions for the screening of employees at the plant, and for design phase review of plant layout and protection of vital equipment have been described. We find that these provisions conform to Regulatory Guide 1.17, "Protection of Nuclear Plants Against Industrial Sabotage." Based on our review, we conclude that the applicant's arrangements for protection of the plant against acts of industrial sabotage are satisfactory for the construction permit stage of review.

14.0 INITIAL TESTS AND OPERATION

The initial test program for the applicant's Cherokee Nuclear Station will be conducted by the applicant with technical direction and support from the nuclear steam supply system vendor (Combustion Engineering) and other vendors, as required. The applicant has committed to develop and execute the test program in accordance with Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors" and Regulatory Guide 1.79 "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors" (Section 6.3.4).

On the basis of our review, we conclude that an acceptable test and startup program can and will be implemented by the applicant. The applicant will provide additional details of this program for our review at the operating license stage of review.

We conclude that the applicant has made acceptable plans for the staffing, development, and conduct of the initial test programs.

712 167

15.0 ACCIDENT ANALYSES

Our evaluation of the capability of the CESSAR standard reference system to withstand abnormal operational transients and postulated accidents is presented in Section 15.0 of Appendix A to this report. The discussion below utilizes the information from Section 15.0 of Appendix A to this report in assessing the radiological consequences of accidents postulated as design basis accidents for the proposed facility.

15.4 Anticipated Transients

Our evaluation of anticipated transients applicable to the proposed facility is presented in Section 15.4 of Appendix A to this report. However, on August 19, 1976, Combustion Engineering presented some experimental results on fuel rod bowing which showed that the plant thermal margins might be less than those intended. Factors that are being considered generically are (1) the gap closure rate for prototypical bundles, (2) the effect on departure from nucleate boiling that bounds the gap closure from part (1), and (3) calculated loss of thermal margin from steps (1) and (2) to reactor transient analyses. An assessment of possible penalties on the proposed facility will be performed during the operating license stage of review.

15.5 Postulated Accidents

15.5.4 Spectrum of Steam Piping Breaks Inside and Outside of Containment

Our evaluation of the steam piping breaks inside and outside of the containment is presented in Section 15.5.4 of Appendix A to this report. The interface requirement that the balance of plant design must satisfy is that the steam line flow restrictors be provided in each line as close as practicable to the steam generator nozzles.

The applicant has committed to incorporate these flow restrictors into the design. This commitment is acceptable for the construction permit stage of review.

15.5.6 Radiological Consequences of Accidents

The postulated design basis accidents analyzed by the applicant to determine the offsite radiological consequences are the same as those analyzed for previously licensed pressurized water reactor plants. These include a design basis loss-of-coolant accident, a steam line break accident, a steam generator tube rupture, a fuel handling accident, a rupture of a radioactive gas storage tank, and a control rod ejection accident. We have reviewed these accidents and have further evaluated the loss-of-coolant accident, and the fuel handling accident.

On the basis of our experience with the evaluation of the steam line break and the steam generator tube rupture accidents for pressurized water reactor plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible reactor coolant system and secondary coolant system radioactivity concentrations. At the operating license stage of review, we will include the limits in the technical specifications on the reactor coolant system and secondary coolant system activity concentrations such that the potential two-hour doses at the exclusion radius, as calculated by the staff for these accidents, will be small fractions of the guideline doses of 10 CFR Part 100. Similarly, we will include limits in the technical specifications on gas decay tank activity so that any single failure, such as the lifting and subsequent failure of a relief valve to close, will not result in doses that are more than a small fraction of the 10 CFR Part 100 guideline values.

Each of the pressurized water reactors for the proposed facility will be surrounded by a double containment structure consisting of a low leakage steel containment vessel and an outer reinforced concrete shield building to minimize the offsite radiological consequences of the design basis loss-of-coolant accident. The applicant has specified a design leak rate for the primary containment of 0.2 percent of containment volume per day for the first 24 hours following the loss-of-coolant accident and 0.1 percent per day for the duration of the accident. For dose evaluation purposes, radioactive materials that leak from the primary containment following a postulated loss-of-coolant accident can take any of the following pathways to the environment:

(1) Leakage to the annulus between the primary and secondary containment structures (the shield building annulus), which will be treated by the annulus ventilation system.

(2) Direct bypass leakage, which will not be treated.

The annulus ventilation system is an engineered safety feature.

The applicant has determined the bypass leakage pathway percentage to be one percent of the total primary containment leakage. We have used this bypass leakage pathway percentage in our calculations of the loss-of-coolant accident doses. The results of our calculations are shown in Table 15.1, and the assumptions used in the analysis are listed in Table 15.2. The doses we [illegible] for the loss-of-coolant accident are within the guideline dose values [illegible]ulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological [illegible] of a Loss of Coolant Accident for Pressurized Water Reactors," for a pla[illegible] [illegible]truction permit stage of review (these limits are 150 rem for the thyroid [illegible]d twenty [illegible] for the whole body).

In modeling the releases through the shield b[illegible] annulus pathway, we assumed that the annulus ventilation system operates at pa[illegible] recirculation throughout the course of the accident following an initial [illegible] pressure transient in the annulus.

712 169

TABLE 15.1

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

Accident	Exclusion Area* 2-Hour Dose, rem		Low Population Zone** 30 Day Dose, rem	
	Thyroid	Whole Body	Thyroid	Whole Body***
Loss-of-Coolant	132	10	4	1
Hydrogen Purge	---	--	< 1	< 1
Fuel Handling	8	3	< 1	< 1
Gas Decay Tank Failure	---	8	--	< 1
Control Rod Ejection Accident	150	4	--	--

*Exclusion area boundary distance = 594 meters

**Low population zone distance = 8000 meters

***Doses from low penetrating beta radiation is considered a skin dose and is not included in the whole body dose

712 170

TABLE 15.2

ASSUMPTIONS USED IN THE CALCULATION OF
LOSS-OF-COOLANT ACCIDENT DOSES

Power Level	4100 thermal megawatts
Operating Time	3 years
Fraction of Core Inventory Available for Leakage	
Iodines	25 percent
Noble Gases	100 percent
Initial Iodine Composition in Containment	
Elemental	91 percent
Organic	4 percent
Particulate	5 percent
Shield Building Annulus Volume Between Upper and Lower Elevation of Shield Building Ventilation System Headers	5.3×10^5 cubic feet
Mixing Fraction in Annulus	50 percent
Primary Containment Leak Rate	
0-24 hours	0.2 percent per day
> 24 hours	0.1 percent per day
Direct Outleakage (No Filtration)	80 seconds
Direct to Atmosphere (Bypass)	0.002 percent per day
Annulus Ventilation System Iodine Filter Efficiencies	
Elemental Iodine	99 percent
Organic Iodine	99 percent
Particulate Iodine	99 percent
Primary Containment Volumes	
Sprayed Volume	2.6×10^6 cubic feet
Unsprayed Volume	6.9×10^5 cubic feet
Containment Spray System Removal Coefficients	
Elemental Iodine	10 per hour
Organic Iodine	0
Particulate Iodine	0.6 per hour
Mixing Rate Between Sprayed and Unsprayed Volumes	2.31×10^4 cubic feet per minute (two turnovers/hour)
Elemental Iodine Decontamination Factor	100
Minimum Exclusion Area Boundary Distance	594 meters
Low Population Zone Distance	8,000 meters

712 171

TABLE 15.2 (continued)

ASSUMPTIONS USED IN THE CALCULATION OF
LOSS-OF-COOLANT ACCIDENT DOSES

Annulus Ventilation Flow Distribution

Time Step	Recirculation Flow (cubic feet per minute)	Exhaust Flow (cubic feet per minute)
0-1 minute	0	0
1-3.5 minutes	4,000	12,000
3.5-4.5 minutes	5,000	11,000
4.5-6 minutes	7,000	9,000
6-8.1 minutes	8,500	7,500
8.1-10 minutes	11,500	4,500
10-16.6 minutes	13,500	2,500
16.6-26.6 minutes	14,800	1,200
26.6 minutes - 1.1 hours	15,200	800
1.1-1.25 hours	15,200	800
1.25-2 hours	15,600	400
2-2.5 hours	15,600	400

Relative Concentration Values (seconds per cubic meter)

0-2 hours @ exclusion area boundary distance (594 meters)	2.5×10^{-3}
0-8 hours @ low population zone distance (8,000 meters)	5.9×10^{-5}
8-24 hours @ low population zone distance (8,000 meters)	3.8×10^{-5}
1-4 days @ low population zone distance (8,000 meters)	1.5×10^{-5}
4-30 days @ low population zone distance (8,000 meters)	4.0×10^{-6}

As part of our evaluation of the loss-of-coolant accident, we have considered the consequences of leakage of containment sump water which is circulated by the emergency core cooling system outside the containment after the postulated accident. We have assumed that the sump water contained a mixture of iodine fission products consistent with the recommendations of Regulatory Guide 1.7 "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident." During the recirculation mode of operation, starting about 2200 seconds after the accident, the sump water is circulated outside of the containment to the reactor building to be cooled. If a source of leakage should develop, such as from a pump seal, a portion of the iodine would become gaseous and would exit to the atmosphere. Since the emergency core cooling system area is served by an engineered safety feature air exhaust filtration system, we conclude that total offsite doses including doses from possible equipment leakage would be within the guidelines of 10 CFR Part 100, even for substantial amounts of equipment leakage.

The applicant will provide redundant hydrogen recombiners for the purpose of controlling any accumulation of hydrogen within the primary containment after a design basis loss-of-coolant accident. In the event both recombiners fail, the applicant has provided a backup purge system which discharges to the shield building annulus and subsequently to the atmosphere through the annulus ventilation system filters. Assuming operation of the annulus ventilation system at full exhaust and with no credit for mixing or holdup in the annulus, we have computed the additional dose an individual might receive due to purging of the containment after the accident. The calculated doses are shown in Table 15.1. The assumptions used in the analysis are listed in Table 15.3. The results of Table 15.1 show that calculated doses at the low population zone resulting from purging, when added to the loss-of-coolant accident doses, are well within the guidelines of 10 CFR Part 100.

A fuel handling accident can occur within containment or within the spent fuel pool area of the auxiliary building. We have not completed our analysis of the accident within containment (Sections 1.9 and 6.2.4). For the spent fuel pool area, we have assumed that a fuel assembly was dropped in the spent fuel pool during refueling operations and that all of the fuel rods in the assembly were damaged, thereby releasing the volatile fission gases from the fuel rod gaps into the pool. The radioactive material that escaped from the fuel pool was assumed to be released to the environment over a two-hour time period with the iodine activity reduced by filtration through the fuel building exhaust system. The dose results are shown in Table 15.1 and the assumptions and parameters used in the analysis are shown in Table 15.4. The dose model and dose conversion factors employed in the analysis were in agreement with those given in Regulatory Guide 1.25, " Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." Calculated doses for the fuel handling accident in the spent fuel pool are well within the guidelines of 10 CFR Part 100. We will report on our evaluation of the radiological consequences for the fuel handling accident within containment in a supplement to this report.

712 173

TABLE 15.3

HYDROGEN PURGE DOSE INPUT PARAMETERS

Power Level	4100 thermal megawatts
Containment Volume	3.3×10^6 cubic feet
Holdup Time in Containment Prior to Purge Initiation	16 days
Purge Duration	30 days
Purge Rate	80 standard cubic feet per minute
Annulus Ventilation System Filter Efficiency for Iodines	99 percent
4-30 days Relative Concentration Value at 8,000 meters	4.0×10^{-6} seconds per cubic meter

TABLE 15.4

ASSUMPTIONS USED IN THE FUEL HANDLING ACCIDENT ANALYSIS

Power Level	4100 thermal megawatts
Number of Fuel Rods Damaged	236
Total Number of Fuel Rods in Core	56,876
Radial Peaking Factor of Damaged Rods	1.65
Shutdown Time	72 hours
Inventory Released from Damaged Rods (Iodines and Noble Gases)	10 percent
Pool Decontamination Factors	
Iodines	100
Noble Gases	1
Iodine Fractions Released from Pool	
Elemental	75 percent
Organic	25 percent
Filter Efficiency for Iodine Removal	99 percent
0-2 Hours Relative Concentration Value @ 594 meters	2.5×10^{-3} seconds per cubic meter

Our evaluation of the radiological consequences of a postulated control rod ejection accident is presented in Section 15.5.6 of Appendix A to this report. There we state that, based on the information supplied in the CESSAR for the control rod ejection accident, we calculated that for a 30-meter elevated release a site with a two-hour relative concentration of 1×10^{-3} seconds per cubic meter or less at the exclusion area boundary is required to meet the 150 rem thyroid dose guideline value. By following the guidance of Standard Review Plan 15.4.8, the two-hour relative concentration value of 5.4×10^{-4} cubic meters per second given by Regulatory Guide 1.5 for a wind speed of one meter per second becomes 1.8×10^{-3} cubic meters per second for the five percentile wind speed of 0.3 meter per second at the site. This relative concentration value of 1.8×10^{-3} seconds per cubic meter could result in a need as stated in Section 15.5.6 of Appendix A to this report to require a reduction in the primary to secondary steam generator tube leak rate from one to 0.55 gallon per minute at the operating license stage of our review in order to meet the 150 rem thyroid dose guideline stated in Appendix A. During our operating license review we will use any additional meteorological data available and will require technical specifications to limit the primary to secondary system steam generator tube leak rate to a value less than the value of one gallon per minute assumed in Section 15.5.6 of Appendix A if necessary to maintain calculated doses below the guideline values of 10 CFR Part 100.

Assumptions used in the calculation of the control element assembly ejection accident doses are tabulated in Table 15.4 in Appendix A of this report. Additional assumptions used but inadvertently not tabulated in that table are:

(9) 1.2 peaking factor

(10) 0.45 percent of the fuel reaches at least incipient centerline melting after the rod ejection accident

(11) 100 percent of noble gases and 50 percent of iodine in fuel reaching incipient centerline melting temperature are released to the primary coolant

15.6 Anticipated Transients Without Scram

As stated in Section 1.10, anticipated transients without scram is an issue that is generic in nature which is being pursued primarily with the vendor in question. Our evaluation of anticipated transients without scram is presented in Section 15.6 of Appendix A to this report. However, subsequent to that review, additional information has been received and is discussed herein.

Concerning the resolution of anticipated transients without scram for the CESSAR design, we requested Combustion Engineering, Inc. to provide the following by June 30, 1976:

(a) The results of additional analysis and further justification of the Combustion Engineering analysis model identified in the staff's Status Report and its supplement.

712 176

(b) Based on these analyses, identification of the design changes needed to assure that the limits specified in WASH-1270 will not be violated following an anticipated transient without scram.

With regard to item (a), Combustion Engineering submitted additional information with supplements to documents CENPD-107, "ATWS Modifications to CESEC," CENPD-135, "STRIKIN-II-a Cylindrical Geometry Fuel Rod Heat Transfer Program," and CENPD-158, "Anticipated Transients Without Reactor Trip." We are now reviewing this information. With regard to item (b), Combustion Engineering proposed to improve the reliability of the CESSAR design shutdown system by modifying the design to include a diverse trip system including the necessary diverse sensor channels, instrument channels, trip logic, and trip actuators. We are now reviewing this proposal. Based on our review, any changes indicated to be needed will be required to be incorporated in the CESSAR System 80 design which is applicable to the proposed facility.

16.0 TECHNICAL SPECIFICATIONS

The technical specifications in an operating license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the Nuclear Regulatory Commission. Final technical specifications will be developed and evaluated at the operating license stage. However, in accordance with Section 50.34 of 10 CFR Part 50, an application for a construction permit is required to include preliminary technical specifications. The regulations require an identification and justification for the selection of those variables, conditions or other items which are determined as a result of the preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, with special attention given for those items which may significantly influence the design.

We have reviewed the proposed technical specifications presented in Section 16.0 of the PSAR with the objective of identifying those items that would require special attention at the construction permit stage, to preclude the necessity for any significant change in design to support the final technical specifications. The proposed technical specifications are similar to those being developed or in use for plants of similar design to the proposed facility. We have not identified any items which require special attention at this stage of our review.

On this basis, we have concluded that the proposed technical specifications are acceptable.

712 178

17.0 QUALITY ASSURANCE

17.1 General

Section 17.0 of the PSAR, which is applicable to the proposed facility references the quality assurance (QA) program description given in the Duke Power Company Topical Report, "Quality Assurance Program - DUKE 1," and Section 17.0 of the CESSAR submitted by Combustion Engineering, Incorporated.

The Duke Power Company is the applicant and engineer-constructor. Combustion Engineering is the supplier of the nuclear steam supply systems.

Our evaluation of the description of the QA program for the proposed facility is based on our review of this information and detailed discussions with the applicant to determine the qualifications and capability of the applicant and the principal contractor (Combustion Engineering) to comply with the requirements of Appendix B to 10 CFR Part 50, applicable Regulatory Guides, and industry standards.

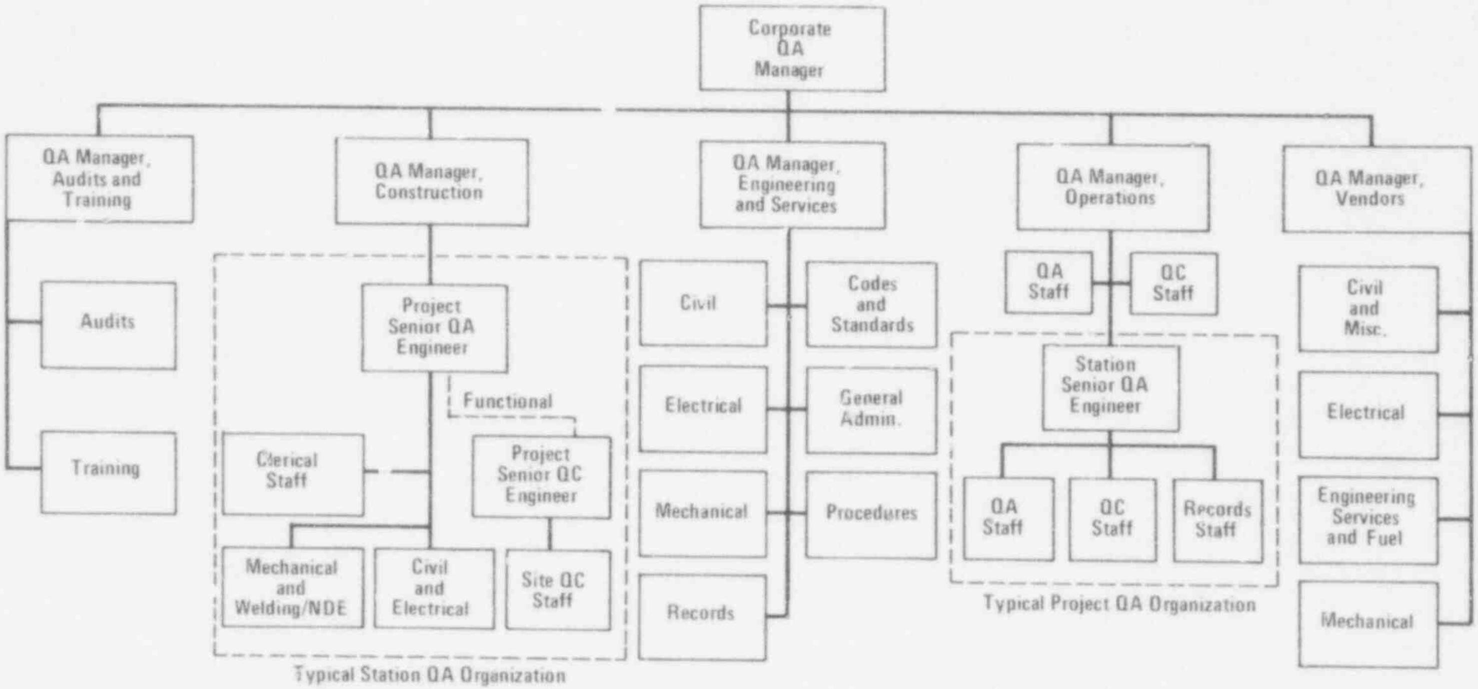
Duke Power Company is responsible for the total Duke Power Company QA program, and is organized to control and verify the QA efforts of Combustion Engineering.

17.2 Duke Power Company

The Duke Power Company includes three major organizational elements reporting to an Executive Vice President and General Manager. One organization is responsible for engineering and construction; one for power generation; and one for purchasing.

The Senior Vice President, Engineering and Construction, is responsible for establishing Duke Power Company's QA policies, goals, and objectives. He has delegated to the Corporate QA Manager, who reports directly to him, the responsibility for managing and implementing this QA program. Duke Power Company's QA Department, under the direction of the Corporate QA Manager, is shown in Figure 17.1. Reporting to the Corporate QA Manager are QA Managers responsible for (1) audits and training, (2) construction, (3) engineering and services, (4) operations, and (5) vendors. The Corporate QA Manager has established procedures, manuals, and instructions for implementing the QA program. The Corporate QA Manager is on the same organizational level as those whose work he verifies. QA Department personnel are organizationally separate and independent from those persons responsible for performing engineering, construction, operational and procurement activities. The QA Department is responsible for design and procurement QA, shop inspections, and witnessing tests.

712 179



Note: This chart shows the total Quality Assurance organization. At various stages during implementation of the new organization or a new nuclear project all positions may not be filled.

Figure 17.1 Duke Power Company Quality Assurance Department

The Construction Department has the responsibility for all site construction activities including field construction testing. A Project Manager, who reports directly to the Vice President, Construction, is assigned to each Duke Power Company project. He is responsible for all site construction activities and onsite quality control (QC) activities. Reporting to the Project Manager are the General Superintendent, who is responsible for all craft activities, including meeting cost and schedule objectives, and the Project Engineer. Reporting to the Project Engineer are the Senior QC Engineer, the Senior Construction Engineer, and the Senior Planning and Facilities Engineer. The Senior QC Engineer, who reports administratively[1] to the Project Engineer and functionally[2] to the Senior QA Engineer, is responsible for inspection on the project.

Our evaluation of Duke Power Company's organizational arrangements for QA and QC is that these are sufficiently independent of the organizations whose activities they verify; they have clearly defined authorities and responsibilities, and are organized such that they can identify quality problems in other organizations performing quality related work; can initiate, recommend, or provide solutions; and can verify implementation of solutions. We therefore conclude that this organizational arrangement complies with Appendix B to 10 CFR Part 50 and is acceptable.

Duke Power Company's original QA program description in the topical report did not provide enough detail to adequately describe the QA program for design, procurement, and construction of its nuclear power plants. In response to our request for a more detailed comprehensive description of the QA program, the Duke Power Company amended its topical report by Amendment Nos. 1, 2 and 3.

The topical report as amended provides a matrix of typical procedures used to administer the QA program along with a brief abstract of the purpose of these and their relationship to the applicable QA requirements of Appendix B to 10 CFR Part 50. Based on our review of this information and other commitments in the topical report, we conclude that each criterion of Appendix B to 10 CFR Part 50 has been acceptably addressed. Further, the structures, systems, and components comprising the safety items which are subject to this program have been identified in the Preliminary Safety Analysis Report.

Duke Power Company's QA program has been developed to conform to the provisions of the Regulatory Guides and industrial standards that are contained in the Commission's documents entitled, "Guidance on QA Requirements During Design and Procurement Phase

[1] "Administrative" means that the Project Engineer has hire/fire, salary review, and work scheduling direction of QA personnel.

[2] "Functional" means that QA has final review and approval of inspection procedures and reports and certification of inspectors.

712 181

of Nuclear Power Plants" (Revision 1), May 24, 1974 (WASH-1283, Revision 1); "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," May 10, 1974 (WASH-1309); and "Guidance on Quality Assurance Requirements During the Operations Phase of Nuclear Power Plants," October 26, 1973 (WASH-1284). Based on this and Duke Power Company's definition of their policies and procedures, we find they have made a satisfactory commitment to an acceptable QA program.

Duke Power Company, by surveillance, will assure that its Engineering and Construction Departments, Combustion Engineering, and key vendors and subcontractors will have adequate QA programs, that inspections will be performed to documented inspection instructions by qualified personnel, and the results will be recorded. Duke Power Company will assure by surveillance and audits that personnel performing inspections are free from undue cost and schedule pressures of the project.

Duke Power Company has developed a formal indoctrination and training program applicable to its personnel, including those in its Design Engineering, Construction, and QA/QC organizations. Quality Control inspectors are, for example, trained and qualified in the specific area in which they will be inspecting. The QA program requires formal training, on-the-job training, examination, and certification of these inspection personnel.

Duke Power Company has established program requirements on itself and on Combustion Engineering and important vendors and subcontractors which assure that there will be a documented system of records attesting to quality.

A system of planned and documented audits, described in the topical report, will be used by Duke Power Company to verify compliance with all aspects of the QA program and to assess its effectiveness. Duke Power Company's audit results will be reviewed and corrective action taken by responsible management. Followup action is taken to assure corrective action. We find that Duke Power Company's audit commitments are strong and well defined.

Duke Power Company's executive level management regularly assesses the scope, implementation, and effectiveness of the QA program by means of project and staff review meetings, by management audits, and by review of trend analyses provided directly to the Senior Vice President, Engineering and Construction.

Based on our review of the description of the QA program contained in Duke Power Company's topical report, we find that there are adequate and well defined procedures, a commitment to the Commission's QA guidance, assurance of an independent inspection program, a documented system of records attesting to quality, an audit system to inform management of the effectiveness of the QA program, and a satisfactory management assessment of the status and adequacy of the QA program.

We conclude that Duke Power Company's QA program described in their topical report on quality assurance, "Quality Assurance Program - DUKE 1," as modified by Amendments 1, 2 and 3 and as referenced in Section 17.0 of the PSAR, includes an acceptable organizational arrangement for QA/QC with adequate policies, procedures, and instructions to implement a program that will satisfy the requirements of Appendix B to 10 CFR Part 50.

17.3 Combustion Engineering, Incorporated

The Combustion Engineering QA program has been evaluated as discussed in Section 17.0 of Appendix A of this report. As noted therein, the QA program for Combustion Engineering does not cover the Combustion Engineering manufacturing work. However, the applicant, in Amendment 23, changed Section 17.1 of the PSAR to state that the QA program described in Section 17.1 of the CESSAR will be followed by the Combustion Engineering manufacturing facilities. The amendment also states that these facilities will meet the applicable portions of WASH-1309. In addition, an organization chart for these manufacturing activities was provided in Section 17.1 of the PSAR.

Based on our review and evaluation of the QA program for Combustion Engineering manufacturing activities as described in Section 17.1 of the PSAR, we have concluded that this QA program demonstrates an acceptable QA organization with adequate policies, procedures, and instructions to implement a program that will satisfy the requirements of Appendix B to 10 CFR Part 50.

17.4 Implementation of the Quality Assurance Program

The Office of Inspection and Enforcement has conducted inspections to examine the implementation of the QA program for the proposed facility. Based on their inspections and assessment, the Office of Inspection and Enforcement concludes that the implementation of PSAR commitments for the proposed facility is consistent with the status of the project.

17.5 Conclusion

In our review, we have evaluated the QA program descriptions of Duke Power Company and Combustion Engineering for compliance with the Commission's regulations and applicable Regulatory Guides and industry standards. Based on this review we conclude that the QA program (1) complies with Appendix B to 10 CFR Part 50 and applicable guides and standards, and (2) is acceptable for the design, procurement, and construction of the proposed facility. The Office of Inspection and Enforcement has concluded that the QA program implementation is consistent with the status of the project and is, therefore, acceptable.

712 183

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The application for the proposed facility is being reviewed by the Advisory Committee On Reactor Safeguards. We intend to issue a supplement to this Safety Evaluation Report after the Committee's report to the Commission relative to its review is available. The supplement will append a copy of the Committee's report and will address the significant comments made by the Committee, and will also describe steps taken by the staff to resolve any issues raised as a result of the Committee's review.

19.0 COMMON DEFENSE AND SECURITY

The applicant states that the activities to be conducted will be within the jurisdiction of the United States and that all the directors and principal officers of the applicant are citizens of the United States.

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 has agreed to safeguard any such data that 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 from military purposes is involved. For these reasons, and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

20.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to financial data and information required to establish financial qualifications for an applicant for a facility construction permit are Paragraph 50.33(f) of 10 CFR Part 50 and Appendix C to 10 CFR Part 50. To assure that we have the latest information to make a determination of the finanacial qualifications of an applicant, it is our current practice to review this information during the later stages of our review of an application. We are continuing our review of the financial qualifications of the applicant and will report the results of our evaluations in a supplement to this report.

21.0 CONCLUSIONS

Based on our analysis of the proposed design of the Cherokee Nuclear Station, Units 1, 2 and 3, and upon favorable resolution of the outstanding matters set forth in Section 1.9 and discussed in appropriate sections of this report, we will be able to conclude that in accordance with the provisions of paragraph 50.35(a) of 10 CFR Part 50:

(1) The applicant has described the proposed design of the facility, including but not limited to the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public;

(2) Such further technical or design information as may be required to complete the safety analysis and which can be reasonably left for later consideration will be supplied in the Final Safety Analysis Report;

(3) Safety features or components which require research and development have been described and identified by the applicant, and there will be conducted research and development programs reasonably designed to resolve safety questions associated with such features or components;

(4) On the basis of the foregoing, there is reasonable assurance that (a) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (b) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facilities can be constructed and operated at the proposed location without undue risk to the health and safety of the public;

(5) The applicant is technically qualified to design and construct the proposed facilities;

(6) The applicant has reasonably estimated the costs and is financially qualified to design and construct the proposed facility; and

(7) The issuance of permits for construction of the facilities will not be inimical to the common defense and security or to the health and safety of the public.

NUREG-75/112
DECEMBER 31, 1975

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

COMBUSTION ENGINEERING, INCORPORATED

CESSAR SYSTEM 80

DOCKET NO. STN 50-470

APPENDIX A

712 188

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

712 195

APPENDICES

LIST OF TABLES

712 197

LIST OF FIGURES

1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

On September 17, 1973, Combustion Engineering, Incorporated (hereinafter referred to as Combustion Engineering) filed with the United States Nuclear Regulatory Commission (the Commission), then known as the United States Atomic Energy Commission, a proposed preliminary reference system design designated as the System 80 design for a nuclear steam supply system. This submittal was in the form of an application for a Preliminary Design Approval by the Commission staff in response to Option 1 of the Commission's standardization policy, WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Plants." Option 1 allows for the review of a "reference system" that involves an entire facility design or major fraction of a facility design outside the context of a license application. The application was docketed on December 19, 1973.

Our review of the CESSAR was similar to our review of a construction permit application, except that it was limited to only those features within the CESSAR scope, plus safety related interfaces between the CESSAR and the balance of plant. Upon completing the review and concluding by the staff and the Advisory Committee on Reactor Safeguards that the design can be implemented without undue risk to the health and safety of the public, a Preliminary Design Approval will be issued rather than a Construction Permit.

The initial Commission policy statement on standardization of nuclear power plants was issued on April 28, 1972. This policy statement provided the impetus to industry and the Commission to initiate active planning in their respective areas. That is, it provided a method whereby the benefits of standardization could be realized while maintaining the Commission's standards for protecting the health and safety of the public and for protecting the environment. On March 5, 1973, the Commission announced its intent to implement a standardization policy for nuclear power plants. In August 1974, the Commission issued its standardization program plan, WASH-1341. Amendment 1 to WASH-1341, dealing with "options" and "overlaps," was issued January 16, 1975. The regulations governing the submittal and review of standard designs under the "reference system" option are found in Appendix O to 10 CFR Part 50 and paragraph 2.110 of 10 CFR Part 2.

A standard safety analysis report entitled "Combustion Engineering Standard Safety Analysis Report" (CESSAR) was submitted with the application. The information in the CESSAR has been supplemented by Amendments 1 through 44. We have completed our review of the CESSAR through Amendment 44. Copies of the CESSAR including these amendments are available for public inspection at the Nuclear Regulatory Commission, Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555.

712 199

On July 3, 1975, we issued a Report To The Advisory Committee On Reactor Safeguards (the Committee) of our evaluation of the CESSAR. This report presented our evaluation of the CESSAR through Amendment 28 for Sections 7 and 8, and through Amendment 29 for all other sections of the CESSAR. On August 8, 1975, we issued Supplement Number 1 to our report to the Committee which presented our evaluation of the CESSAR through Amendment 29 for Sections 7 and 8, and through Amendment 34 for all other sections of the CESSAR. On August 14, 1975, the Committee considered the application, and on September 17, 1975 issued its report to the Commission. A copy of the Committee's report is attached as Appendix C, and the results of the Committee's review are discussed in Section 18 of this report.

This Safety Evaluation Report summarizes the results of the technical evaluation of the proposed System 80 design performed by the Commission staff, and delineates the scope of the technical matters considered in evaluating the radiological safety aspects of the System 80 design. This report also addresses the comments by the Committee in its report of September 17, 1975 and the resolution of outstanding issues previously identified in our report to the Committee. The environmental aspects of the CESSAR were not considered in the review; however, they will be addressed for each site-related application which references the CESSAR.

Based on our evaluation of the CESSAR, we conclude that the proposed preliminary design of the nuclear steam supply system can be incorporated by reference in construction permit and standard balance of plant design applications and can be constructed without endangering the health and safety of the public. We conclude that a Preliminary Design Approval for the proposed design can be granted. Our detailed conclusions are presented in Section 19 of this report.

As stated previously, Combustion Engineering is responsible for the design of those systems within the CESSAR System 80 design scope. Applicants for construction permits for plants incorporating the System 80 design will retain contractors such as architect-engineers, constructors, turbine-generator vendors, and consultants as needed. We will need to conclude for each such application that the selected site is acceptable and that the applicant and relevant contractors are technically competent to manage, design, construct and operate a specific reactor plant incorporating the System 80 design prior to issuing a Construction Permit.

The review and evaluation presented in this report is the first stage of a continuing review by the Commission staff of the design, construction, and operating features of the System 80 design. Prior to the issuance of an operating license for any application incorporating or referencing the CESSAR we will review the final design of the CESSAR System 80 reference system to determine that all of the Commission's safety requirements have been met in accordance with our regulations. The expected end product of our review of the final design of the CESSAR reference system would be a Final Design Approval, rather than an Operating License.

In the course of our safety review of the material submitted, we held numerous meetings with Combustion Engineering representatives to discuss the designs of the systems proposed in the CESSAR, and their performance under normal, transient and postulated accident conditions. During the course of our review, we have requested Combustion Engineering to provide additional information for our evaluation. This additional information was provided in amendments to the CESSAR.

As a result of our review, numerous changes were made in the nuclear steam supply system design. These changes are described in the amendments, and discussed in appropriate sections of this report. A chronology of the principal actions relating to the processing of the application is attached as Appendix A to this report. Our bibliography for this report is attached as Appendix B.

1.2 General Description

The proposed System 80 reference system will consist of a pressurized water reactor with a two loop reactor coolant system and the auxiliary systems directly related with the nuclear steam supply system as illustrated in Figure 1.1. In keeping with the guidelines of Regulatory Guide 1.49 (Revision 1), "Power Levels of Nuclear Power Plants," the CESSAR is an application for a Preliminary Design Approval for a core thermal power of 3800 megawatts. The proposed System 80 will be housed in a containment building not within the scope of the CESSAR, which will be designed by the balance of plant architect engineer or by the utility-user that incorporates the System 80 design. The scope of the proposed reference System 80 design will include only those systems and components which are directly related with the normal operation and emergency shutdown of the reactor.

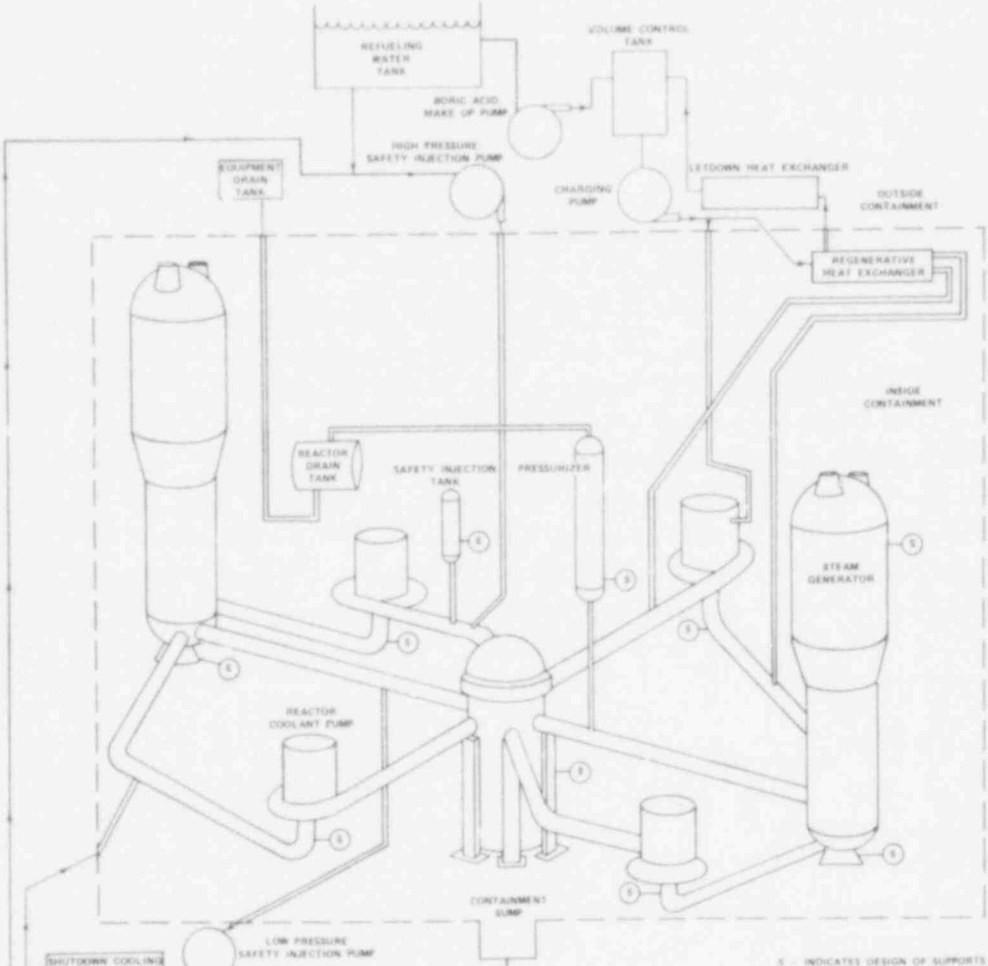
The System 80 nuclear steam supply system is a design for a single unit. Systems and components within the nuclear steam supply system that are important to safety will not be shared.

In addition, although the CESSAR scope does not include conventional balance of plant features such as auxiliary service facilities and general service facilities (e.g., the site, plant buildings and structures, the ultimate heat sink, onsite and offsite electrical systems, the main steam system excluding the steam generators, and the turbine-generator and its auxiliaries), the CESSAR scope does include the delineation of interface requirements pertaining to those balance of plant features that have a direct bearing on the integrity or on the functional capability of the safety related systems within the CESSAR scope.

The proposed reference System 80 initially contained certain optional features that could be elected, at the option of the balance of plant designer or utility user that utilizes the design. These options were subsequently deleted. The reference System 80 in the CESSAR will consist of the following systems:

(1) Reactor system

712 201



S - INDICATES DESIGN OF SUPPORTS

NOTE: THIS FIGURE IS NOT INTENDED TO SHOW ALL INTERCONNECTIONS BETWEEN SYSTEMS

CESSAR SYSTEMS NOT ILLUSTRATED:
- REACTOR PROTECTION SYSTEM
- ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS
- FUEL HANDLING SYSTEM

Figure 1.1 - CESSAR Design Scope

(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

1.2.1 Reactor System

The proposed pressurized water reactor system will include the reactor vessel, a standard design of integral supports, reactor vessel head cover, the reactor core and all internal appurtenances required to support the reactor core. The reactor core will be composed of uranium dioxide pellets enclosed in Zircaloy-4 tubes with welded end plugs. The fuel tubes will be grouped and supported in assemblies. The reactor core will initially be loaded in three regions. All fuel in each region will have the same enrichment of uranium-235, which will differ from the enrichment used in the other regions.

1.2.2 Reactor Coolant System

The reactor coolant system will consist of two closed reactor coolant loops. Each loop will include a steam generator and two reactor coolant pumps. Water will both moderate and cool the core. The water will be circulated through the reactor vessel and core and two reactor coolant loops by four reactor coolant pumps. The water heated by the reactor will flow through the two steam generators where heat will be transferred to the secondary (steam) system, and then back to the reactor through the pumps to complete the cycle. An electrically heated pressurizer with a safety valve system will be connected to one of the reactor coolant loops to establish and maintain reactor coolant pressure. The major components of the reactor coolant system will incorporate standard designs of integral supports and snubbers. These supports will be provided for the steam generators, the reactor coolant pumps, and the pressurizer.

1.2.3 Reactor Control System

The reactor will be controlled by two reactivity control systems: (1) control element assemblies, the vertical movement of which will compensate for or initiate rapid changes in reactivity; and (2) dissolved boron, the adjustment of concentration of

712 203

which will compensate for long term variations in reactivity due to fuel burnup and fission product concentration changes, and to ensure ample shutdown margin during refueling.

The vertical movement of the control element assemblies will be accomplished by magnetic jack type drives (control element drive mechanisms). The concentration of boron will be adjusted by the chemical and volume control system.

1.2.4 Reactor Protective System

The reactor protective system will consist of sensors, calculators, logic circuits, and related supporting equipment to monitor selected nuclear steam supply system conditions. Redundancy, diversity, independence and separation of reactor protective circuits will be provided in accordance with the Commission's criteria. Four measurement channels will be provided for each monitored parameter connected in a two-out-of-four logic matrix for a reactor trip signal. The reactor trip signal will, in turn, cause the coils of the control element drive mechanisms to be deenergized, thereby releasing the control element assemblies so that they may drop into the core.

1.2.5 Engineered Safety Features Actuation System

The engineered safety features actuation system within the CESSAR System 80 reference scope will consist of the electrical and mechanical devices and circuitry, from the sensors to the actuation device input terminal.

1.2.6 Chemical and Volume Control System

The purity, volume and boric acid content of the reactor coolant will be controlled by the chemical and volume control system. The purity will be controlled by continuous purification of a bypass stream of reactor coolant in the chemical and volume control system. The level in the reactor coolant system pressurizer will be automatically controlled by varying the amount of coolant discharged (letdown) and the amount pumped back into the system by the charging pumps. Boron concentration will be controlled by a feed and bleed method whereby the purified letdown stream will be diverted to a boron recovery section of the chemical and volume control system, from where either concentrated boric acid or demineralized water will be added to the reactor coolant via the charging pumps.

1.2.7 Shutdown Cooling System

During plant shutdown operations, the reactor coolant system temperature will be reduced from the normal operating temperature to about 350 degrees Fahrenheit by venting the steam generator to the turbine condenser or to the atmosphere if the turbine condenser is not available. The shutdown cooling system will be provided to cool the reactor coolant system from 350 degrees Fahrenheit down to a cold shutdown or

712 204

refueling temperature. The shutdown cooling system will cool the reactor coolant by utilizing the low pressure safety injection pumps to circulate the reactor coolant through the shutdown cooling heat exchangers.

1.2.8 Safety Injection System

A safety injection system (emergency core cooling system) will be provided as part of the engineered safety features system to localize, control, mitigate and terminate postulated accidents, including a loss-of-coolant accident. The safety injection system will include four safety injection tanks, and independent and redundant low pressure and high pressure safety injection trains designed to automatically inject highly borated water into each of the four reactor coolant system cold legs. This system will assure core cooling and protection for the complete range of postulated primary and secondary coolant pipe break sizes.

1.2.9 Fuel Handling System

A fuel handling system will be provided for the safe handling of fuel assemblies and control element assemblies for refueling or maintenance purposes. This system will provide for the assembly, disassembly and storage of the reactor vessel head and internals, and will include: (1) a refueling machine, (2) a fuel transfer carriage, (3) tilting machines, (4) a fuel transfer tube, (5) a spent fuel handling machine in the fuel handling building, and (6) various devices used for handling the reactor vessel head and internals.

1.3 Comparison with Similar Designs

Many features of the CESSAR System 80 design are new Combustion Engineering designs, and some aspects of the plant are similar to those that we have previously evaluated and approved for other nuclear power plants. Our review of the CESSAR has, therefore, to the extent feasible and appropriate, made use of our previous evaluations of features that are similar to those in the CESSAR.

To assist in understanding the relationship of the System 80 design to other Combustion Engineering designs, a comparison of the principal design features of San Onofre Units 2 and 3 (Docket Nos. 50-361 and 50-362) and those of the CESSAR System 80 design is presented in Table 4.1. Our Safety Evaluation Reports for San Onofre Units 2 and 3 and other applications using Combustion Engineering nuclear steam supply system designs are available for public inspection in the Nuclear Regulatory Commission Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555.

1.4 Requirements for Future Technical Information

Section 1.5 of the CESSAR describes test programs that Combustion Engineering will conduct to demonstrate the safety of the CESSAR System 80 design. These programs and their objectives are listed in Table 1.1 of this report.

TABLE 1.1

COMBUSTION ENGINEERING TEST PROGRAMS

TEST	PURPOSE OF TEST
1. 16 x 16 Fuel Assembly Design Tests	
Upper Guide Structure and Control Element Assembly Buffer Test	Verify structural and functional adequacy of the control element assembly guide tube structure buffer design.
Components Proof Test	Verify scram characteristics, scram time and fuel uplift forces, and proof test the control element assembly, control element drive mechanism, guide structure and fuel assembly.
Spacer Grid Test	Verify structural characteristics.
Fuel Assembly Static Test	Verify lateral load deflection.
Fuel Assembly Dynamic Test	Verify pluck, pluck impact, vibratory and axial impact effects.
Reactor Flow Model Test	Verify design hydraulic parameters.
Departure From Nucleate Boiling Improvement Test	Verify thermal performance capability.
Incore Flow Mixing Test	Verify rate of intersubchannel energy transfer due to turbulent interchange and flow scattering of coolant.
2. Fuel Development Programs	
Densification Program	Verify effects of fuel processing methods and parameters on in-reactor densification at high linear power and burnup.
3. Loss-of-Coolant Accident Refill and Blowdown Heat Transfer Tests	
Loss-of-Coolant Accident Refill Tests	Verify the capability of the emergency core cooling system to recover the core after a loss-of-coolant accident.
Blowdown Heat Transfer Test	Verify Dougall-Rosenow correlation, and the transient critical heat flux and the post-critical heat flux heat transfer coefficients.
4. Reflood Test	Verify the reflood heat transfer coefficients.
5. Iodine Decontamination Test	Verify Combustion Engineering's assumed iodine partition factors as described in CENPD-67.
6. Iodine Spiking Test	Develop a realistic and conservative model for the iodine spiking phenomenon.
7. Steam Generator Program	Verify the analytical models used to predict transient and accident loads on the steam generator.
8. Core Protection Calculator Program	Demonstrate the performance of the proposed core protection calculator system software and hardware.

Combustion Engineering did not initially propose reflood heat transfer tests as part of the 16 x 16 fuel design verification program. The reflood heat transfer characteristics of the 16 x 16 fuel are expected to differ from those of the 14 x 14 design, and reflood heat transfer data for the specified Combustion Engineering 16 x 16 design are not available; therefore, we requested that Combustion Engineering provide confirmatory reflood heat transfer test results for a similar design. In response to our request, Combustion Engineering has committed in the CESSAR to: (1) pursue a combined analytical and experimental program directed at establishing and verifying appropriate reflood heat transfer coefficient data for the 16 x 16 fuel assembly design, and (2) submit confirmatory data from this program in the System 80 Final Safety Analysis Report. We find these commitments acceptable for the Preliminary Design Approval stage of our review.

All test programs listed in Section 1.5.2 of the CESSAR that are related to development of the System 80 fuel assembly are scheduled to be completed by the end of 1976; however, fuel fabrication for CESSAR System 80 reactors is not scheduled to start until 1979. Thus, the results of the test programs should be available prior to completion of the 16 x 16 core design, thereby allowing ample time for core design changes if any of the test programs produce unexpected results. Irrespective of the time available for any required modification, we conclude that the commitment and requirement to provide results of analyses, tests, and surveillance of the System 80 fuel assembly design prior to Final Design Approval are acceptable for the Preliminary Design Approval stage of our review.

CESSAR Section 1.5.3 outlines the iodine decontamination factor test program that Combustion Engineering has undertaken to substantiate its position on radioiodine partitioning in the steam generator. This test program is described in Combustion Engineering Topical Report CENPD-67, "Iodine Decontamination Factors During PWR Steam Generation and Steam Venting," dated September 1973. Combustion Engineering has supplemented CENPD-67 with Revision 1, dated November 1974, to incorporate additional data obtained from the test program. We have reviewed the Combustion Engineering information and conclude that the decontamination factor should be divided into two parts, and that separate factors should be used for inorganic and organic iodine. Additionally, Combustion Engineering has committed (Section 1.5.3.3. of the CESSAR) to supplement CENPD-67 with operating plant data, test apparatus development verification data obtained, and any conclusions. We conclude that this commitment is acceptable for the Preliminary Design Approval stage of our review.

CESSAR Section 1.5.4 describes an iodine spiking test program that Combustion Engineering has proposed. This test program is in response to our requirement that iodine spiking be considered in determining the source terms that are used in evaluating steam line and steam generator tube rupture accidents. The test program will consist of high sampling rate of primary coolant at an operating nuclear power plant during shutdown operations to determine iodine isotope concentrations as a function of time. Combustion Engineering will use this data to derive an iodine spiking model. Until such time as an acceptable model is developed, we require that the currently available data be applied to provide a conservative estimate of iodine spiking in evaluating the balance of plant design and in assessing site characteristics.

CESSAR Section 1.5.5 describes a steam generator development program that Combustion Engineering has proposed to confirm its evaluation of the structural integrity of the proposed System 80 steam generator design. The development program will address, in particular, the effect of the integral economizer design during thermal transients and during main steam line or feedwater line break accidents. On the basis of our review of the information provided in the CESSAR and discussions with Combustion Engineering concerning this program and our review of the steam generator design and operation as described in CESSAR Section 5.5.2, we conclude that the proposed steam generator development program can reasonably be expected to provide a basis for substantiating the models used to evaluate the dynamic loads on the steam generator. Combustion Engineering will report on the results of this program in a topical report. We will require that Combustion Engineering adequately demonstrate, in this topical report, that the models being used to evaluate dynamic loads are adequately conservative. Combustion Engineering has committed (CESSAR Section 1.5.5.4) to submitting the topical reports by December 1976 which will include all experimental data and substantiate structural integrity of the steam generator under operational and accident transients.

CESSAR Section 1.5.6 describes Combustion Engineering's proposed development program for the core protection calculator system. We have concluded that the proposed program can reasonably be expected to determine the adequacy of the proposed core protection calculator system design described in CESSAR Appendix 7A. The design and test program for this system is being evaluated under a generic review (see Section 7.2), the results of which will be required to be available in time to permit an alternate design to be implemented, if necessary. In the highly unlikely event that the development program results show the proposed core protection calculator system design to be unacceptable, the alternate design will be implemented. CESSAR Section 1.5.6 identifies an alternate design for implementing the departure from nucleate boiling ratio and local power density protection functions that are expected to be provided by the core protection calculator system. The alternate design is that provided for the Florida Power and Light St. Lucie Unit No. 1 (Docket No. 50-335), which has been reviewed and found to be acceptable.

1.5 Summary of Principal Review Matters

Our evaluation of the systems designs proposed in the CESSAR included a technical review of the information submitted by Combustion Engineering, particularly with regard to the following principal matters:

(1) We evaluated the design and expected performance characteristics of the proposed nuclear steam supply system described in the CESSAR to determine whether the safety related systems conform with the Commission's General Design and Quality Assurance Criteria, and applicable guides, codes and standards. We also evaluated

the design to determine whether all departures from criteria, codes and standards have been identified and justified.

(2) We evaluated the expected response of the proposed System 80 reference system design to various anticipated operating transients, and to a broad spectrum of postulated accidents, and determined that the potential consequences of a few postulated, but highly unlikely, accidents (design basis accidents) would exceed those of all other accidents considered. We performed conservative analyses of these design basis accidents to determine if the potential offsite doses that might result from these accidents would be well within the Commission's guidelines for site acceptability, as given in 10 CFR Part 100, for typical sites when the CESSAR System 80 design is mated with an acceptable balance of plant design.

1.6 Resolution of Outstanding Issues

In our report to the Advisory Committee on Reactor Safeguards dated July 3, 1975, we identified certain outstanding issues which required that Combustion Engineering provide additional information to confirm that the proposed design would meet our requirements, or where our review was not yet complete. We have resolved all these issues in a manner acceptable for issuance of a Preliminary Design Approval. These items are discussed further in applicable sections of this report.

1.7 Interface Information

Although the CESSAR does not cover the entire facility, it does specifically describe or delineate the safety-related interface requirements imposed on the balance of plant design by the CESSAR System 80 design. These interfaces include seismic design response spectra, dimensional and structural requirements, operating environment input to transient and accident analysis, and the performance requirements necessary to assure compatibility of the CESSAR System 80 with the mating portions of the plant and site. Although the CESSAR is not associated with any particular site, representative site parameters have been assumed and used in sample dose calculations by Combustion Engineering and the Commission staff.

In an effort to develop a consistent and reasonable policy for handling interfaces, we have held numerous staff meetings, and have met on numerous occasions with standard plant applicants.

At the time of issuance of our Report to the Advisory Committee on Reactor Safeguards on the CESSAR System 80 design, we had determined that the interface information provided by Combustion Engineering through Amendment 28 was inadequate. Accordingly, we embarked upon a joint program with Combustion Engineering in order to establish acceptable interfaces for the CESSAR System 80 design. The program included an update of the interface information provided in the CESSAR to Amendment 26, and an audit by the staff of the engineering information normally transmitted by Combustion

Engineering to its utility customers and their architect-engineers. The results of our audit and review of the updated interface information were used to identify additional interface information that we required. Combustion Engineering has provided this information in subsequent amendments to the CESSAR, up through Amendment 44.

We have completed our review of the interface information provided by Combustion Engineering through Amendment 44 and have determined that this information is sufficient to determine the compatibility of the safety-related systems and components within the scope of the CESSAR System 80 design with the balance of plant design to be submitted in applications referencing the CESSAR. The interface information provided in the CESSAR is also adequate to determine the validity of the CESSAR System 80 accident analyses when the CESSAR System 80 is referenced by a balance of plant design application. We therefore conclude that the CESSAR System 80 interface information is acceptable for Preliminary Design Approval purposes.

2.0 SITE CHARACTERISTICS

Appendix O to 10 CFR Part 50 requires that standard design applications shall include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such postulated site parameters. Although the CESSAR does not address specific site locations nor specific site parameters, it does contain interface information for certain site related design bases. Specifically, these interfaces are:

(1) Seismic Considerations -- The seismic design response spectra curves given in Section 3.7.1 of the CESSAR define the seismic limitations for reactor coolant system major component supports, nozzles, and piping, and represent the envelope of actual design requirements for current plants. These limiting design response spectra form an envelope which will exceed the seismic severity for most sites in the Continental United States.

(2) Other Natural Phenomena Considerations -- The interface requirements for the protection of safety related equipment from site related hazards such as winds and tornadoes, floods, and missiles are discussed in Sections 3.3.3, 3.4.5, and 3.5.4, respectively, of the CESSAR.

We find the site related interface information provided in the CESSAR acceptable, and the assumptions used in the CESSAR sufficiently conservative or representative of sites for Preliminary Design Approval purposes. However, each utility-user referencing the CESSAR must show that the site related safety parameters for each specific site are within the design envelope of the CESSAR.

3.0 DESIGN CRITERIA FOR SYSTEMS AND COMPONENTS

3.1 Conformance with the General Design Criteria

Combustion Engineering has presented its evaluation of the design bases for the System 80 reference system, with respect to the Commission's General Design Criteria as contained in Appendix A of Part 50 to Title 10 of the Code of Federal Regulations (10 CFR Part 50), in CESSAR Section 3.1. Based on our evaluation of the preliminary design and of the proposed design criteria, we conclude that the design of the nuclear steam supply system set forth in the CESSAR is in conformance with the Commission's General Design Criteria.

3.2 Classification of Systems and Components

3.2.1 Seismic Classification

Criterion 2 of the General Design Criteria 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 functions. 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 the guideline exposures of 10 CFR Part 100.

We have reviewed the seismic classification of System 80 fluid systems and components important to safety that are within the scope defined in the CESSAR and will be designed to withstand, without loss of function, the effects of a safe shutdown earthquake. These fluid systems and components are: (1) reactor coolant system, (2) safety injection system, and (3) the safety-related portions of the chemical and volume control system. Excluded from this review are structures and balance of plant fluid systems that interface with System 80 fluid systems. The safety class and seismic classification of the balance of plant structures, systems and components will be reviewed for each user's application.

We have reviewed the auxiliary systems for the reactor coolant pumps, including their lubricating oil system and cooling water system. The lubricating oil system for the reactor coolant pumps will be designed to seismic Category I requirements in accordance with our recommendations. We, therefore, conclude that the design of the lubricating oil system for the reactor coolant pumps is acceptable.

Combustion Engineering has submitted a topical report on the loss of component cooling water to the System 80 reactor coolant pumps, CENPD-201, "Performance of C-E System 80

Reactor Coolant Pump with Loss of Component Cooling Water." In this report Combustion Engineering has identified the pump pivoted pad thrust bearing as the most heavily loaded of any of the bearings in the pump-motor assembly. This bearing also presents the most metal to metal surface area which if allowed to come in contact could produce the most friction and thus must effect pump coastdown capabilities. In Appendix A of CENPD-201, a calculation demonstrates that the bearing assembly can function without component cooling water for a period of time in excess of 30 minutes. Within this time period, the calculated increase in the sump oil temperature is acceptable and provides assurance that the lubrication oil will maintain a film clearance between the metal surfaces of the bearing. No seizure of the pump shaft is predicted to occur and the pump coastdown rate would be unaffected by the increase in sump oil temperature. We find that Combustion Engineering has demonstrated that there is sufficient time available within which an operator can trip the reactor coolant pumps and initiate a safe plant shutdown. Loss of component cooling water to the pump seal assembly would have little effect on the seals since primary cooling of the seals is provided by the precooled seal injection flow from the chemical and volume control system. These seal injection lines are classified seismic Category 1 and Quality Group B.

In the event Combustion Engineering should utilize the reactor coolant pumps to mix borate solution required to bring the reactor to a cold shutdown condition or to residual heat removal system operating conditions in accordance with the plant technical specifications, this should be accomplished prior to pump shutdown. The technical specifications for a plant incorporating the System 80 design will require that: (1) the reactor coolant pumps be shut down 30 minutes after loss of component cooling water, (2) the reactor coolant pumps will not be restarted until component cooling water is restored and pump thermal conditions are normal, and (3) prior to reactor coolant pump shutdown a sufficient amount of boron will be introduced into the reactor coolant system to facilitate cooldown to residual heat removal system operating conditions. Since there is no available operating experience with System 80 reactor coolant pumps, Combustion Engineering will need to perform a reactor coolant pump test to verify the analysis in Appendix A of CENPD-201 with regard to the performance of the pump-motor thrust bearings during a loss of component cooling water to the cooling coils in the sump oil reservoir. This test will be performed on a prototype pump from a nominal initial temperature until the temperature of the lubricating oil reaches a value which permits the conservatism in the analysis to be assessed. For this purpose, sump temperatures in the range of 200 degrees Fahrenheit would be tested. During this period, component cooling water to the pump shaft seal assembly will also be terminated.

Our acceptance of the proposed classification of the component cooling water lines to the pump seal assembly, and the pump-motor thrust bearings and motor exit air coolers as Quality Group D and designed to non-seismic Category I requirements is based on the commitment by Combustion Engineering to demonstrate by test the capability of the System 80 reactor coolant pump to perform during the required pump test without component cooling water as defined above. We will evaluate the results of the test during our review of the application for Final Design Approval. If the results of the test are not acceptable, we will require that the cooling water system for the pump seals and pump-motor thrust bearings be designed to seismic Category I requirements.

Systems, and components important to safety that will be designed to withstand the effects of a safe shutdown earthquake and remain functional, have been identified in an acceptable manner and classified as seismic Category I items in conformance with Regulatory Guide 1.29, "Seismic Design Qualification," in Table 3.2-1 of the CESSAR. All other systems, and components that may be required for operation of the nuclear steam supply system, are designed to other than seismic Category I requirements. Included in this classification are those portions of Category I systems which will not be required to perform a safety function.

The basis for acceptance in our review has been conformance of Combustion Engineering's design, design criteria and design bases for systems and components important to safety with the Commission's regulations as set forth in Criterion 2 of the General Design Criteria, and to Regulatory Guide 1.29, and industry codes and standards.

We conclude that System 80 systems and components important to safety will be designed in accordance with seismic Category I requirements which provide reasonable assurance that in the event of a safe shutdown earthquake, the plant will perform in a manner providing adequate safeguards to the health and safety of the public.

3.2.2 System Quality Group Classification

Criterion 1 of the General Design Criteria requires that nuclear power plant systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

We have reviewed Combustion Engineering's classification system for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves in fluid systems important to safety, and the assignment by Combustion Engineering of safety classes to those fluid systems required to perform a safety function.

Combustion Engineering has applied the classification system of the American Nuclear Society (Safety Classes 1, 2, 3 and 4), which corresponds to the Commission's Quality Groups A, B, C and D in Regulatory Guide 1.26, "Quality Group Classification and Standards," to those fluid containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety. Reliance is placed on these systems to (1) prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) permit shutdown of the reactor and maintenance in the safe shutdown condition, and (3) contain radioactive material. Fluid system pressure-retaining components important to safety that are classified Quality Groups A, B or C will be constructed to the ASME Boiler and Pressure Vessel Code as follows:

Quality Group	Component Code Section III, Division 1, 1974 Edition
A	Class 1
B	Class 2
C	Class 3

Quality Group A components will comply with Section 50.55a of 10 CFR Part 50. Quality Groups B and C components will comply with Subsection NA-1140 of the code.

Components that are classified Quality Group D will be constructed to the following codes as appropriate: ASME Boiler and Pressure Vessel Code, Section VIII, Divisions 1 or 2, and ANSI B31-1-1973. Quality Group D components such as orifices, boron meters, strainers and gas traps will be constructed to no code.

The System 80 fluid systems identified in Section 3.2.1 have been classified in an acceptable manner in CESSAR Table 3.2-1 and on system piping and instrumentation diagrams, in conformance with Regulatory Guide 1.26. As noted in Section 3.2.1, excluded from this review are those structures and balance of plant fluid systems that interface with System 80 fluid systems.

The basis for our acceptance has been conformance of Combustion Engineering's designs, design criteria, and design bases for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping and valves in fluid systems important to safety, with the regulations as set forth in Criterion 1 of the General Design Criteria, the requirements of the Codes specified in Section 50.55a of 10 CFR Part 50, Regulatory Guide 1.26, and industry codes and standards.

We conclude that System 80 fluid system pressure-retaining components important to safety that are designed, fabricated, erected and tested to quality standards in conformance with these requirements provide reasonable assurance that the plant will perform in a manner providing adequate safeguards to the health and safety of the public.

3.3 Wind and Tornado Loadings

Design provisions for protection of the System 80 reference system against the effects of winds and tornadoes will be discussed in each user's application. CESSAR (Amendment 39) includes an interface requirement that the location, arrangement, and installation of systems and components required for safe plant shutdown shall be such that the effects of winds and tornadoes will not prevent these systems and components from performing their shutdown functions. We conclude that the CESSAR, as amended, provides the necessary information for Preliminary Design Approval purposes with respect to wind and tornado loadings of the systems described in the CESSAR, and therefore is acceptable. A detailed evaluation of each user's application will be performed to ascertain that wind and tornado loadings have been appropriately considered.

712 215

3.4 Water Level (Flood) Design

Design provisions for protection of the System 80 reference system against the effects of floods will be discussed in each user's application. CESSAR (Amendment 36) includes an interface requirement that the location, arrangement, and installation of systems and components required for safe plant shutdown shall be such that the effects of floods (including tsunami and seiches for applicable sites) will not prevent these systems and components from performing their shutdown functions. We conclude that the CESSAR, as amended, provides the necessary information for Preliminary Design Approval purposes with respect to flood design of the systems described in the CESSAR, and therefore is acceptable. A detailed evaluation of each user's application will be performed to ascertain that the effects of floods have been appropriately considered.

3.5 Missile Protection Criteria

Criterion 4 of the General Design Criteria requires that systems and components important to safety be protected against the effects of missiles generated both from within the reactor building (internally generated missiles) and external to the reactor building. The responsibility for protection of safety-related systems and components is not within the scope of the CESSAR. Our review, therefore, was limited to identifying the sources of internally generated missiles, and verifying that appropriate interface requirements are included to protect the safety-related systems in the CESSAR System 80 design from missiles.

We have reviewed the systems and components to be protected from missiles, potential missile sources associated with component overspeed failures of equipment within the System 80 scope, and missiles that could originate from high pressure system ruptures of equipment and systems within the scope of the CESSAR System 80 design.

Design provisions for protection of the System 80 reference system against the effects of missiles will be discussed in the balance of plant designer's or utility-user's application. CESSAR Section 3.5.4.1 provides design criteria and interface requirements for systems and components inside and outside containment which require that appropriate design features such as missile barriers, natural separation, and orientation be provided to insure that the impact of any potential missile will not lead to a loss of coolant accident, or preclude systems from carrying out their specified safety functions, or prevent the plant from remaining in a safe shutdown condition. We have reviewed the interface requirements in the CESSAR and conclude that they are acceptable with respect to missile protection. The protection afforded against internally generated missiles outside containment will be evaluated for all applications referencing the CESSAR System 80 design.

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

Criterion 4 of the General Design Criteria requires that structures, systems and components important to safety shall be appropriately protected against the dynamic effects from the postulated rupture of piping.

We reviewed the proposed System 80 design to determine that the design will accommodate the effects of postulated pipe breaks and jet impingement from piping systems. For systems located inside containment, the CESSAR is consistent with Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," with respect to criteria that will be employed for determination of (1) the systems which will be evaluated, (2) the locations and types of piping breaks which will be postulated, and (3) the protective measures against pipe-whip that will be provided.

The analytical methods and procedures that will be used to determine the most probable type of pipe break at a particular location, the pipe motion subsequent to rupture, and the pipe-whip restraint dynamic interaction appropriately consider the structural characteristics of the system.

The provisions for protection against the dynamic effects associated with pipe ruptures and the resulting discharging coolant provide acceptable assurance that, in the event of the occurrence of the combined loadings imposed by an earthquake of the magnitude specified for the safe shutdown earthquake and a concurrent single pipe break of the largest pipe at any one of the design basis break locations, the following conditions and safety functions will be accommodated and assured:

(1) The magnitude of the design basis loss-of-coolant accident cannot be aggravated by potential multiple failures of piping.

(2) The reactor emergency core cooling systems can be expected to perform their intended function.

(3) Systems and components important to safety will be appropriately protected.

On the basis of the above findings, we conclude that the criteria that will be used for the identification, design, and analysis of piping systems where postulated breaks may occur inside containment constitute an acceptable design basis in meeting the applicable requirements of the Commission's General Design Criteria 1, 2, 4, 14 and 15.

Combustion Engineering has committed that the design of high and moderate energy systems which are part of the basic nuclear steam supply system located outside containment will be in accordance with the guidance set forth in Mr. J. O'Leary's letter dated July 12, 1973. Design basis piping breaks for high and moderate energy are postulated to occur in any branch or run of piping larger than 1 inch nominal diameter. Combustion Engineering has provided, as interface information, the temperature and pressures for high and moderate energy fluid systems outside containment within the scope of the CESSAR System 80 design. The CESSAR identifies essential systems and components to be protected from piping failures outside containment.

Based on the information and commitments in the CESSAR, we conclude that the design criteria for high and moderate energy systems outside containment are in accordance with our guidelines, and that applicants referencing the CESSAR System 80 design can develop an acceptable design so that postulated pipe breaks in System 80 systems installed outside containment will not prevent the safe shutdown of the reactor. We will evaluate each user's application as to the detailed implementation of these criteria.

3.7 Seismic Design

3.7.1 Seismic Input

Criterion 2 of the General Design Criteria 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 functions.

Seismic response spectra have been provided for the reactor coolant system major component supports, nozzles and piping, and for auxiliary and fuel handling equipment. In any application where the CESSAR is incorporated or referenced, it is required that the response spectra for the actual site conditions and structures for the plant be within the envelope of the response spectra in the CESSAR.

Combustion Engineering will provide the following interface information to assist the balance of plant designer in the design of supports and related structures.

(1) A description of the seismic response spectra envelopes at all support points and the maximum relative displacement between support points for which System 80 components are designed.

(2) Envelopes of the seismic loads transmitted from Category I or non-Category I systems connecting to CESSAR System 80 system components.

(3) A simplified mathematical model which accounts for the mass and stiffness properties of systems within System 80 which can be coupled with the mathematical model of the seismic system including structures and supports.

The specific percentages of critical damping values proposed to be used in the analysis of seismic Category I equipment and components are in conformance with Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," and are considered conservative for use in the seismic analysis, and therefore acceptable. We will evaluate each user's application to ensure that the response spectra in the CESSAR envelop the response spectra for the actual site and plant structures.

3.7.2 Seismic System Subsystem Analysis

Our review of the seismic system and subsystem analysis for CESSAR System 80 included the analysis methods for seismic Category I systems and components, including the review of procedures for modeling, methods of analysis and criteria for incorporating the three directional seismic motion.

The system and subsystem analyses will be performed on an elastic basis. Modal response spectrum multi-degree-of-freedom and time history methods will form the bases for the analysis of all major seismic Category I systems and components. When the modal response spectrum method is used, governing response parameters will be combined by the square root of the sum of the square rule. However, the absolute sum of the modal responses will be used for modes with closely spaced frequencies. The square root of the sum of the squares of the maximum codirectional responses will be used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. A vertical seismic system dynamic analysis will be employed for all seismic Category I components and equipment where analyses show significant structural amplification in the vertical direction. Torsional effects will also be considered.

We determined that the proposed seismic system and subsystem analysis procedures and criteria will assure conservative predictions for seismic loads, and, therefore, conclude that the proposed procedures and criteria provide an acceptable basis for the seismic design.

3.8 Design of Seismic Category I Structures

Plant structures are not within the scope of CESSAR System 80. However, the System 80 design interfaces with the balance of plant seismic Category I structures are discussed in the CESSAR. Combustion Engineering will provide the utility-users of the CESSAR the following interface information:

(1) The maximum allowable differential displacements due to all loads (normal, thermal, seismic, ...) at points of the nuclear steam supply system that will interface with balance of plant structures.

(2) The structural properties (e.g., support stiffnesses) of the supporting balance of plant structures that were used in the analysis of the nuclear steam supply system and which must be satisfied.

(3) All the loads that have to be transmitted from the nuclear steam supply system components to the supporting balance of plant structures.

(4) The limitations on deflections of the balance of plant structures supporting the nuclear steam supply system components under all loading conditions.

We conclude that these interface requirements are adequate to provide acceptable protection for the nuclear steam supply system.

3.9 Mechanical Systems and Components

3.9.1 Dynamic Analysis and Testing

3.9.1.1 Piping Vibration Operational Test Program

In accordance with the American Society of Mechanical Engineers (ASME) Code, Section III, paragraphs NB-3622.3 and NC-3622, which require that the designer be responsible by observation under startup or initial operating conditions for ensuring that the vibration of piping systems is within acceptable levels, utility-users for nuclear power plants incorporating or referencing the CESSAR will be required to conduct a piping vibration operational test program. For such plant applications we will review the preoperational dynamic effects program that will be conducted on all ASME Code Class 1 and Class 2 piping systems and related restraints, components, and supports during startup and during the initial operating conditions testing to verify that the programs are acceptable.

The CESSAR provides a preoperational piping test program that covers the reactor coolant loop and surge line piping only; therefore, the testing program appropriate to all other piping will be provided in each application referencing the CESSAR. Specifically, the guidelines to determine where and how the visual observations will take place, including the methods and procedures to determine whether the observed vibration intensity is excessive, will be provided in each Preliminary Safety Analysis Report referencing the CESSAR.

The testing programs will develop loads similar to those experienced during reactor operation and will be consistent with the Commission's Standard Review Plan, Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment".

We conclude that the test programs will provide adequate assurance that the piping and piping restraints of the systems are designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design operational transients and will constitute an acceptable basis for partial fulfillment of the requirement established by the Commission's General Design Criterion 15.

3.9.1.2 Analysis and Testing of Mechanical Equipment

The dynamic testing and analysis procedures which will be implemented to confirm (1) that all seismic Category I mechanical equipment within the CESSAR System 80 scope of design will function during and after an earthquake of magnitude up to and including the safe shutdown earthquake, and (2) that all equipment support structures are adequately designed to withstand seismic disturbances, are acceptable.

Subjecting the equipment and its supports to dynamic testing and analysis procedures provides reasonable assurance that, in the event of an earthquake at the site, the seismic Category I mechanical equipment, as identified in the CESSAR, will continue to function during and after a seismic event, and the combined loading imposed on the equipment and its supports will not exceed applicable code allowable design stress and strain limits. Limiting the stresses of the supports under such loading combinations provides an acceptable basis for the design of the equipment supports to withstand the dynamic loads associated with seismic events, as well as operational vibratory loading conditions, without loss of structural integrity.

712 220

Implementation of these dynamic testing and analysis procedures constitutes an acceptable basis for satisfying the applicable requirements of the Commission's General Design Criteria 2 and 14.

3.9.1.3 Reactor Internals Flow-Induced Vibration Testing

With regard to flow-induced vibration testing of reactor internals for the System 80 units, the first System 80 plant to be ready for hot functional tests will be the prototype plant and will be tested in accordance with Regulatory Guide 1.20, "Vibration Measurements on Reactor Internals." For subsequent System 80 plants, additional confirmatory vibration testing and monitoring programs with subsequent visual inspection will be conducted to provide added confirmation of the capability of the structural elements of the reactor internals to sustain flow-induced vibrations. The programs will be consistent with Regulatory Guide 1.20.

We have reviewed the preoperational vibration test programs proposed by Combustion Engineering for verifying the design adequacy of the reactor internals for both the prototype and non-prototype System 80 plants, under loading conditions that will be comparable to those experienced during operation. We conclude that the combination of tests, predictive analysis, and post-test inspection will provide adequate assurance that the reactor internals can be expected to withstand flow-induced vibrations without loss of structural integrity during their service lifetime. Prior to issuing an operating license, we will review the preoperational vibration test program that will be performed in accordance with Regulatory Guide 1.20, for each utility-user application, for assurance that it constitutes an acceptable basis for demonstrating the design adequacy of the reactor internals and satisfying the applicable requirements of the Commission's General Design Criteria 2 and 14.

3.9.1.4 Correlation of Test and Analytical Results

The correlation of tests and analytical results will be discussed in each user's application.

3.9.1.5 Analysis Methods for Loss-of-Coolant Accident Loadings

Each user referencing the CESSAR will perform a dynamic system analysis of the reactor internals and of the broken and unbroken piping loops. The purpose of this analysis will be to provide acceptable bases for confirming: (1) the structural design adequacy of the reactor internals, and (2) that the unbroken piping loops can withstand the combined dynamic effects of the postulated occurrence of a loss-of-coolant accident and a safe shutdown earthquake.

We have reviewed the analytical methods proposed in the CESSAR for performing the dynamic system analysis and find that use of these methods will assure that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction as specified in Appendix F to the ASME Boiler and Pressure Vessel Code, Section III, and that the resulting deflections or displacements of any structural elements of the reactor internals will not distort the geometry of the reactor internals to the extent that core cooling can be impaired.

The assurance of structural integrity of the reactor internals under the postulated safe shutdown earthquake and the most severe loss-of-coolant accident conditions provides added confidence that the design can be expected to withstand a spectrum of lesser pipe breaks and seismic loading combinations.

We conclude that the use of the proposed analytical techniques will assure an acceptably conservative structural design for the System 80 reactor internals, and constitutes an acceptable basis for satisfying the requirements of the Commission's General Design Criteria 2 and 4.

On May 7, 1975, we were informed by a licensee of a pressurized water reactor, Virginia Electric and Power Company, that an asymmetric loading resulting from a postulated pipe rupture at a particular location in the reactor coolant system had not been taken into account in the original design of the reactor pressure vessel support system for the North Anna Units 1 and 2 (Docket Nos. 50-338 and 339). This loading results from the forces induced on the internals within the reactor vessel caus... differential pressure conditions within the vessel immediately following a post...ted loss-of-coolant accident. In addition, the asymmetric loading from transient differential pressures that would exist around the exterior of the reactor vessel from the same postulated pipe rupture was not included in the original design analysis. However, the symmetric loadings from such a postulated pipe rupture were included in the original analysis of the reactor pressure vessel supports.

It is our opinion that these factors related to the design of the reactor pressure vessel supports are generic in nature and may apply to the CESSAR System 80 design. Accordingly, we are taking steps to review this problem on a generic basis to determine the extent of the problem.

We have informed Combustion Engineering of the nature of this problem and requested verification that the design procedures for the reactor pressure vessel support system will properly include the asymmetric forces described above in the final design of the supports. In a letter dated December 2, 1975, Combustion Engineering provided verification that the final design will include the asymmetric forces. Combustion Engineering stated that a dynamic analysis will be performed using a lumped parameter model including details of the reactor vessel and supports, major connected piping and components, and the reactor internals. The pipe break thrust force, asymmetric subcompartment pressurization forces and asymmetric reactor internals hydraulic forces will be applied as simultaneous time history forcing function. We conclude that the use of the proposed techniques will assure a conservative design for the reactor vessel supports and constitutes an acceptable design basis for the Preliminary Design

Approval. We require that the details of this analysis be submitted for our review prior to the Final Design Approval.

3.9.2 ASME Code Class 2 and 3 Components

All safety-related ASME Code Class 2 and 3 systems, components, and equipment will be designed to sustain normal loads, anticipated transients, the operating basis earthquake and the safe shutdown earthquake within design limits which are consistent with those outlined in Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components." The specified design basis combinations of loadings as applied to the design of the safety-related ASME Code Class 2 and 3 pressure-retaining components in systems classified as seismic Category I provide reasonable assurance, that in the event an earthquake should occur at the site, or other upset, emergency or faulted plant transients should occur during normal plant operation, the resulting combined stresses imposed on the system components may be expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without loss of structural integrity. The design load combinations and associated stress and deformation limits proposed for all ASME Code Class 2 and 3 components constitute an acceptable basis for design in satisfying the Commission's General Design Criteria 1, 2, and 4.

The criteria that are proposed for use in developing the design and mounting of ASME Class 2 and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices will provide a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function.

The criteria proposed for the design and installation of ASME Class 2 and 3 overpressure relief devices constitute an acceptable design basis in meeting the applicable requirements of the Commission's General Design Criteria 1, 2, 4, 14 and 15 and are consistent with those specified in Regulatory Guide 1.67, "Installation of Overpressure Protection Devices."

In connection with analytical and empirical methods for the design of pumps and valves, Combustion Engineering has described an operability qualification program for active safety-related Class 2 and 3 pumps and valves. This program is in agreement with the Commission's Standard Review Plan, Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

Implementation of the proposed operability assurance program will provide adequate assurance of the capability of active pumps and valves in seismic Category I systems including those which may be classified as ASME Code Class 1, 2 and 3 to withstand postulated seismic loads in combination with other significant loads without loss of structural integrity, and to perform the "active" function (i.e., pump operation, valve closure or opening) when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated. The proposed component operability assurance procedures constitute an acceptable basis for meeting the requirements of General Design Criteria 1, 2 and 4 as related to operability of ASME Code Class 1, 2 and 3 active pumps and valves.

We have reviewed the requirements for fracture toughness testing and properties proposed by the applicant to provide assurance that the pressure-retaining ferritic materials of all Class 2 and Class 3 components (outside as well as within the reactor coolant system) will have adequate toughness under test, normal operation, and transient conditions. The pressure-retaining ferritic materials of all Class 2 and 3 components, including vessels, will satisfy the requirements of paragraphs NB-2332 and NB-2311 of the Summer 1972 Addenda to Section III of the ASME Code.

The fracture toughness tests and properties required by the Summer 1972 Addenda to Section III of the ASME Code provide reasonable assurance that safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for the pressure-retaining ferritic materials of all Class 2 and 3 components, both within and outside the reactor coolant system.

3.9.3 Standardized Plant Design Interface for Mechanical Components, Systems and Testing Procedures

Combustion Engineering's and the balance of plant designer's responsibilities for mechanical components, systems and testing procedures have been appropriately identified. Combustion Engineering has provided commitments to furnish necessary interface information to the balance of plant designer in accordance with our interface requirements. We find these commitments acceptable.

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

Instrumentation and electrical components required to perform a safety function will be designed to meet seismic Category I design criteria. Seismic requirements established by the seismic system analysis will be incorporated into equipment specifications to assure that the equipment purchased or designed will meet seismic requirements equal to or in excess of the requirements for seismic Category I components, either by appropriate analysis or by qualification testing.

Implementation of the proposed seismic qualification program for seismic Category I instrumentation and electrical equipment and the associated supports for this equipment will provide assurance that such equipm[illegible] expected to function properly and that the structural integrity of the suppo[illegible] ..ll not be impaired during the excitation and vibratory forces imposed by the safe shutdown earthquake and the conditions of post-accident operation. The proposed program will constitute an acceptable basis for satisfying the applicable requirements of the Commission's General Design Criterion 2.

Seismic Category I instrumentation and electrical equipment will be qualified in accordance with requirements and equipment specifications that are consistent with Institute of Electrical and Electronic Engineers (IEEE) Standard 344-1971, "IEEE Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations," and the guidance of Enclosure 2 to our Request for Additional Information, dated March 29, 1974, entitled "Electrical and Mechanical Equipment Seismic Qualification Program." Conformance with these documents will provide acceptable methods for seismic qualification of instrumentation and electrical equipment.

3.11 Environmental Design of Mechanical and Electrical Equipment

Our evaluation of the environmental design of mechanical and electrical equipment is discussed in Section 7.6.1 of this report.

4.0 REACTOR

4.1 Summary

The System 80 reactor design presented in the CESSAR is similar to that reviewed and approved for San Onofre Units 2 and 3 (Docket Nos. 50-361 and 362); however, it includes numerous changes from previous Combustion Engineering designs including San Onofre Units 2 and 3. The significant changes are:

(1) Use of fuel assemblies with a 16 x 16 fuel rod array rather than 14 x 14 fuel rod array.

(2) A thermal power level of 3800 megawatts compared to 3390 megawatts for San Onofre Units 2 and 3.

(3) More fuel assemblies and a larger vessel diameter.

(4) The control element assembly design provides for attaching 4, 8 or 12 control rods to each control element assembly. The rods from a single control element assembly may be inserted into several different fuel assemblies, rather than having only 4 rods per control element assembly that can be inserted into only one fuel assembly. The control rods for the 16 x 16 fuel rod array are slightly smaller in diameter than those in the 14 x 14 array, and the poison used is boron carbide rather than boron carbide-indium-cadmium.

(5) A calandria structure has been added between the fuel alignment plate and the lower base plate of the upper guide structure. The calandria provides individual shroud tubes for each of the control rods that provide lateral support and guidance for the control rods. The calandria design necessitates the removal of all of the control rods when the calandria is removed to gain access to the fuel assemblies. This results in reactor refueling operations being performed without any control element assemblies in the core.

(6) Bottom-entry, fixed and movable in-core neutron flux detectors have replaced the usual top entry fixed position detectors. This modification required the addition of instrument nozzles to the bottom head and guide tubes inside the lower plenum of the reactor vessel.

A comparison of the System 80 and San Onofre Units 2 and 3 core mechanical, nuclear, and thermal hydraulic design parameters is given in Table 4.1.

TABLE 4.1

REACTOR DESIGN COMPARISON

MECHANICAL DESIGN PARAMETERS	SYSTEM 80[1]	SAN ONOFRE UNITS 2 & 3
Number of Fuel Assemblies in Core	241	217
Number of Control Element Assemblies (full length/part length)	81/8	61/8
Number of Fuel Rod Locations in Core[2]	56,876	38,192
Fuel Pellet Diameter, inches	0.325	0.3795
Cladding Inner Diameter, inches	0.332	0.388
Cladding Outer Diameter, inches	0.382	0.44
Cold Diametral Gap, inches	0.007	0.0085
Fuel Pellet Density, percent theoretical value	95	93
Fuel Pellet Enrichment, percent uranium-235		
Region A	1.9	1.9
Region B	2.4	2.3
Region C	2.9	2.9
Fuel Rod Pitch, inches	0.5063	0.58
Active Fuel Length, inches	150	150
Number of Spacer Grids	12	9
THERMAL AND HYDRAULIC PARAMETERS		
General Characteristics		
Total Core Heat Output		
thermal megawatts	3,800	3,390
million British thermal units per hour	13,000	11,600
Heat Generated in Fuel Rod, Core Fraction		
Core Average	0.965	0.975
Hot Rod	0.96	0.975
Pressure, pounds per square inch, absolute	2,250	2,250
Coolant Inlet Temperature, degrees Fahrenheit	565	553
Vessel Outlet Temperature, degrees Fahrenheit	621	611
Core Bulk Outlet Temperature, degrees Fahrenheit	623	613
Total Primary Coolant Flow, million pounds per hour	164	147.8
External Leakage, percent	4.0	3.7
Coolant Flow through Core, million pounds per hour	157.4	142.6
Hydraulic Diameter Nominal Channel, feet	0.0394	0.04445
Core Flow Area, square feet	60.8	53.2
Core Average Mass Velocity, million pounds per hour per square foot	2.59	2.68
Average Coolant Velocity In-Core, feet per second	16.6	16.8
Core Average Heat Flux, British thermal units per hour per square foot	182,200	205,100
Total Heat Transfer Area, square feet	69,000	55,000

TABLE 4.1 (Cont'd)

THERMAL AND HYDRAULIC PARAMETERS	SYSTEM 80[1]	SAN ONOFRE UNITS 2 & 3
Film Coefficient at Average Conditions, British thermal units per hour, per square foot, per degree Fahrenheit	6,300	6,160
Average Film Temperature Difference, degrees Fahrenheit	30	34
Average Linear Heat Rate of Rod, kilowatts per foot (for fraction generated in average rod)	5.53	6.92
Specific Power, kilowatts per kilogram of uranium	38.3	35.7
Power Density, kilowatts per liter	95.6	94.7
Average Core Enthalpy Rise, British thermal units per pound	82.4	81.1
Heat Flux Factors		
Total Nuclear Peaking Factor, (F_Q^N)	2.28 (2.09)*	2.52
Engineering Heat Flux Factor	1.03	1.03
Total Heat Flux Factor, (F_Q^T)	2.35 (2.15)*	2.60
Rod Radial Nuclear Factor	1.55	1.55
Engineering Factor on Linear Heat Rate	1.03	1.03
Enthalpy Rise Factors		
Heat Input Factors		
Design Nuclear Enthalpy Rise Factor	1.51	1.5
Engineering Factor on Hot Channel Heat Input	1.03	1.03
Total Heat Input Factor	1.55	1.55
Rod Pitch, Bowing and Clad Diameter	1.05	1.05
Total Enthalpy Rise Factor (ratio of hot channel enthalpy change to core enthalpy change)	1.67	1.68
Hot Channel and Hot Spot Parameters		
Maximum Heat Flux, British thermal units per hour, per square foot	425,700	553,000[3]
Maximum Linear Heat Rate of Rod, kilowatts per foot (for fraction generated in hot rod at 102 percent rated power)	13.3 (12.1)*	18[3]
Uranium Dioxide Temperature, Steady State, Maximum During Fuel Life, degrees Fahrenheit	3,420	4,010[3]
Maximum Clad Surface Temperature, degrees Fahrenheit	656	657[3]
Hot Channel Outlet Temperature, degrees Fahrenheit	653	646[3]
Hot Channel Outlet Enthalpy, British thermal units per pound	704	687[3]
Departure from Nucleate Boiling Ratio (Modified W-3 Correlation), Steady State	2.22	2.11

*Values resulting from revised loss-of-coolant accident analyses (see Section 6.3).

712 228

TABLE 4.1 (Cont'd)

Table Notations

(1) Parameters based on eight burnable poison rods per fuel assembly.

(2) In the first core, some uranium dioxide rods will be replaced by burnable poison rods.

(3) These parameters are revised as indicated in Supplement 1 to CENPD-46, "Analyses on Combustion Engineering 3410 MWt Plant Emergency Core Cooling System Performance in Accordance with AEC Interim Acceptance Criteria," July 7, 1972.

4.2 Mechanical Design

4.2.1 Fuel

The System 80 fuel assembly design includes 236 fuel rod positions arranged in a 16 x 16 array. Each assembly has five guide tubes, four are for the control rods of the control element assembly which will be located symmetrically, and the other guide is for in-core instrumentation which will be located in the center of each fuel bundle. Each guide tube will occupy the space of four fuel rods. Fuel pellets of 95 percent dense uranium di :ide will be sealed in Zircaloy-4 tubing and pressurized with helium to form the fuel rods. These rods will be positioned with Zircaloy-4 spacer grids of the leaf-spring type. Neutron absorber (poison) rods will be provided in place of fuel rods at selected locations in the fuel assemblies of the first core. The neutron absorber material will be boron carbide dispersed in alumina pellets that are clad in Zircaloy to form rods similar to the fuel rods.

The System 80 fuel assembly design (16 x 16) will be mechanically similar to previously reviewed Combustion Engineering fuel assemblies (14 x 14), such as for San Onofre Units 2 and 3, except that the fuel assembly will be provided with a threaded joint. The threaded joint will allow for attaching the upper and lower end fittings to the guide tubes so they may be removed to allow replacement of individual fuel rods. Other major differences between the 16 x 16 and 14 x 14 designs are summarized in Table 4.1. These differences are essentially geometric and result in a lower linear power density of the fuel rods which increases the thermal performance safety margins.

Evaluation of the Combustion Engineering System 80 fuel design is based upon engineering analyses, mechanical tests, and in-reactor operating experience. Additionally, the performance of the design will be subject to continuing surveillance of operating reactors by Combustion Engineering and individual utilities. These programs continually provide confirmatory and current design performance information.

One of the major thermal analysis considerations reviewed by the staff is related to fuel densification. The initial density of the fuel pellets and the size, shape and distribution of pores within the fuel pellet influence the densification phenomenon. The effects of densification on the fuel rod will increase centerline temperature and the stored energy, increase the linear thermal output, increase the probability for local power spikes (augmentation), and the potential for cladding collapse.

Combustion Engineering has conducted an extensive study of fuel densification and has developed a conservative time-dependent description of the densification process which is described in Combustion Engineering Topical Report CENPD-118, "Densification of Combustion Engineering Fuel," dated June 1974. These densification kinetics along with data on fuel swelling, thermal expansion, fission gas release, fuel relocation, thermal conductivities, cladding creep, and other properties, have been combined in a detailed fuel performance evaluation model which is presented in Combustion Engineering Topical Report CENPD-139, "Combustion Engineering Fuel Evaluation Model," dated July 1, 1974. This model is used to calculate fuel temperature and stored energy, changes in linear thermal output and augmentation (power spike) factors. We have reviewed CENPD-139 and concluded that the fuel performance evaluation model is a generically acceptable method of describing fuel behavior, as discussed in our acceptance letter to Combustion Engineering, dated December 4, 1974, and is applicable to the System 80 fuel. There are several reasons for applicability of the generic model: (1) The specific fuel fabrication process is tied to the densification model through resintering tests which are used to determine the amount of incore fuel densification, and (2) the thermal performance computer code is compared with a body of experimental data whose design parameters include those of the Combustion Engineering fuel.

Although the System 80 fuel has been demonstrated to densify very little and, therefore, should not be prone to form axial gaps between the fuel pellets during densification, it has not been conclusively shown that axial gaps will not appear. We will, therefore, require that an analysis of the clad collapse with a postulated axial gap in the fuel column be performed. Combustion Engineering has submitted to the staff a computer code which will calculate time-to-collapse of Zircaloy cladding in a pressurized water reactor environment, CENPD-187, "CEPAN Method of Analyzing Creep Collapse of Oval Cladding." The staff is in the process of reviewing this computer program. Meaningful calculations of time-to-collapse require actual fabrication data. These calculations, therefore, are not warranted at this time, and the information reviewed is acceptable for Preliminary Design Approval. The time-to-collapse calculations will be reviewed for the Final Design Approval.

We have reviewed the information provided in the CESSAR concerning Combustion Engineering's experience with burnable poison rods similar to those proposed for System 80 and have concluded that the proposed design of the burnable poison rods is acceptable for Preliminary Design Approval purposes. Several Combustion Engineering plants have operated for one cycle with burnable poison rods. In some cases burnable poison rod length increased more than fuel rod length. This will be corrected for System 80 fuel by geometric changes of the pellets. This new design is undergoing irradiation proof testing in core two of Maine Yankee (Docket No. 50-309). The data will be available in early 1977, well before System 80 fuel fabrication.

Combustion Engineering will perform a complete and detailed mechanical design analysis of the System 80 fuel rod and fuel assembly that will specify the materials properties, design loads, limits and associated margins over the whole range of temperatures and burnups expected during the life of the fuel. The analysis will include the effect of shipping, handling, normal operation and postulated accidents on the rods and fuel assembly. The results of Combustion Engineering's analysis are to be submitted in the application for a Final Design Approval; at that time we will verify that the design will provide acceptable safety margins and acceptable mechanical performance of the fuel rods and fuel assembly in shipping, handling, normal operation, and in postulated accidents.

Combustion Engineering will provide a topical report in early 1976 that will describe the methodology to be used in analyzing the combined effect of a loss-of-coolant accident and a seismic event on the fuel assembly. The analytical methods to be used by Combustion Engineering to assess the combined effects of the loss-of-coolant accident and seismic event on the System 80 fuel assembly, and the results of analyses that demonstrate the adequacy of the fuel assembly design are to be submitted in the application for a Final Design Approval.

Mechanical tests to demonstrate the effects of flow-induced vibration and consequent fretting corrosion have been performed on test assemblies and on full size (14 x 14) fuel assemblies to demonstrate that flow vibration induced fretting or wear is acceptably low. Similar full scale hot flow testing of 16 x 16 assemblies will be performed to substantiate these results for the new 16 x 16 design. Combustion Engineering has stated (Section 4.3.1.3.5) that component flow testing will be completed in early 1976, well before the fuel fabrication date for System 80 fuel. The results of these tests are to be submitted in the application for a Final Design Approval.

Testing is currently in progress to provide verification of the structural integrity of the new removable end fitting to guide tube joint design. This testing is described in Section 4.2.1.3 of CESSAR. We will require that the adequacy of this design be demonstrated by these or other acceptable tests prior to our final approval of the proposed System 80 fuel assembly design. The evaluation of this design, supported by the test data, is to be submitted in the application for a Final Design Approval.

Combustion Engineering has described its proposed program of in-reactor and out-of-reactor experiments to demonstrate acceptable performance of the System 80 fuel design in Section 1.5.2 of the CESSAR. In addition, Combustion Engineering has committed to place some fuel assemblies in the Arkansas Nuclear One, Unit 2 reactor (Docket No. 50-368), which will be precharacterized to establish baseline data which can be used to analyze dimensional changes in the fuel assemblies during irradiation. These data will be necessary to quantify any dimensional changes which might occur in the fuel assembly during irradiation such as might occur due to fuel rod bowing. The staff will monitor these inspections to assure that the fuel is performing as expected.

The programs proposed by Combustion Engineering in Section 1.5.2 of the CESSAR, and the characterization and analysis of some of the fuel assemblies following use in a reactor such as the Arkansas Nuclear One, Unit 2, together with the other tests discussed above are necessary and will adequately demonstrate the performance of the System 80 fuel design, if they are supplemented by a fuel surveillance program in at least two plants using Combustion Engineering reactors with the 16 x 16 fuel assembly core. The surveillance program should provide visual inspection of all the peripheral rods in one hundred percent of the initial fuel assemblies, once they are moved from the core to the fuel pool. We will require that such a program be performed in two of the first plants to use a core load of Combustion Engineering 16 x 16 fuel assemblies, and that an evaluation of the results of this program to demonstrate the adequacy of the System 80 fuel design be provided for our review. At the present time it is expected that the two plants will be the Arkansas Nuclear One, Unit 2 plant and the St. Lucie, Unit 2 plant. This supplemental surveillance program will be a proof test to give final reassurance that no long term detrimental behavior has occurred. While it will not have the advantage of being capable of quantifying effects, it will be a thorough survey of the entire core.

During our review, we requested that Combustion Engineering provide certain information that it is unable to provide at the present time, but has committed to provide as discussed above. We will require that this information be submitted for our review, prior to our final approval of the use of the proposed System 80 fuel design in an operating plant. The information presented in the CESSAR, as augmented by the additional analyses, tests and surveillance discussed above, will demonstrate the acceptability of the proposed fuel design. We therefore conclude that, augmented by the additional commitment and requirement to provide analyses, tests and surveillance, the proposed System 80 fuel design is acceptable for the Preliminary Design Approval stage of our review.

4.2.2 Reactor Pressure Vessel Internals

The materials for construction of components of the reactor vessel internals have been identified in Table 4.2-5 of the CESSAR and found to be in conformance with the requirements of Sections II and III of the ASME Boiler and Pressure Vessel Code. All the materials that will be used in the reactor vessel internals conform to the requirements of Appendix I of Section III. The materials selected have been used extensively in industrial applications under similar and more severe conditions with satisfactory results. Past performance of those materials in nuclear power plant applications also has been satisfactory. The proposed controls on the reactor coolant chemistry further ensure that these materials will be adequately protected during reactor operation from an environment which could lead to stress corrosion cracking of the materials and loss of component structural integrity.

Addition of hydrazine and ammonia to the demineralized reactor coolant water in order to scavenge oxygen and increase the alkalinity is a recommended procedure. This recommendation minimizes the potential of halide-induced corrosion of the materials comprising the reactor vessel internals, which could occur if significant quantities of either chlorides or fluorides were present in combination with dissolved oxygen in the reactor coolant. The hydrazine decomposes to form ammonia at higher temperatures. The resultant increase in alkalinity aids in the development of passive oxide films on the surfaces of the reactor coolant pressure boundary. It has been established that the corrosion rates of nickel-chromium-iron Alloy-600 and 300 series stainless steel decrease with time when exposed to the reactor coolant treated in this manner, approaching low steady state values within 200 days. The high alkalinity produced by ammonia addition minimizes corrosion product release and assists in the rapid development of passive films.

The proposed controls on the fabrication of austenitic stainless steel components to be used in the reactor coolant pressure boundary satisfy the recommendations of Regulatory Guides 1.31, "Control of Stainless Steel Welding," and 1.44, "Control of the Use of Sensitized Stainless Steel." Materials selection, fabrication practices, examination procedures, and protection procedures that will be performed in accordance with these recommendations provide reasonable assurance that the austenitic stainless steel used for reactor internals will be in a metallurgical condition which precludes susceptibility to stress corrosion cracking during service.

Use of materials proven to be satisfactory by actual service experience and conformance with the recommendations of Regulatory Guides 1.31, and 1.44, constitutes an acceptable basis for meeting the requirements of General Design Criteria 1 and 14.

Martensitic stainless steel is specified in the CESSAR for some reactor internals. The staff requires a special heat treatment for these materials as follows. After quenching or normalizing from 1775-1825 degrees Fahrenheit, a minimum tempering temperature of 1125 degrees Fahrenheit for 4 hours should be used in order to minimize susceptibility to stress corrosion cracking of the material. Heat treatment of martensitic stainless steel, as specified in the CESSAR, will comply with the staff's position, and is acceptable.

4.2.3 Reactivity Control System

Reactor power will be controlled by movement of control rods and by varying the concentration of a soluble chemical neutron absorber (boric acid).

The proposed rod control system consists of 81 clusters of full-length rods and 8 clusters of part length rods. The full-length rods will be automatically positioned by the reactor control system to shape the reactor power distribution, and to compensate for changes in reactivity resulting from fuel burnup; however, the part-length rods will move only in response to operator action to control axial neutron flux shape and axial xenon oscillations should they occur.

Each cluster will have either 4, 8 or 12 absorber rods. A rod cluster control assembly will comprise a group of individual neutron absorber rods fastened at the top end to a common spider assembly. The absorber material used in the control rods will be boron carbide which is black to thermal neutrons and has additional absorption capability for epithermal neutrons.

The full length rod cluster control assemblies will be divided into two groups, control and shutdown. The control group will compensate for reactivity changes due to variations in operating conditions of the reactor, i.e., power and temperature variations. The insertion of both control and shutdown control element assemblies is required to maintain shutdown margin.

712 234

The concentration of the soluble chemical neutron absorber (boric acid) will be varied through use of the chemical and volume control system to control the slow reactivity changes that result from (1) fuel depletion and fission product buildup, (2) the cold to hot, zero power reactivity change, (3) intermediate term fission product buildup such as xenon and samarium, and (4) burnable poison depletion. Our evaluation of the chemical and volume control system is discussed in Section 9 of this report.

Combustion Engineering used a rod drop time of 3.3 seconds for 90 percent reactivity insertion in the safety analyses. This drop time is based on previous measurements in a 14 x 14 fuel rod bundle using five finger control element assemblies. Similar rod drop tests for the System 80 design will be performed as part of the verification test program. Should these tests warrant a change in the rod drop time, the revised drop time will be used to assess the adequacy of the design at the Final Design Approval stage of review.

Scram tests will be performed utilizing the twelve fingered control element assembly, control element drive mechanism and buffer for various aligned and misaligned conditions for the reactor internals. Scram characteristics and pressure buildup in the buffer region will be measured using the reference design buffer. If necessary, buffer development tests using variations in clearances and lengths will be performed until the desired design characteristics are obtained.

In addition to the above, Combustion Engineering has committed to seismic qualification of all seismic Category I mechanical equipment within the scope of the CESSAR System 80 design to confirm that such equipment will function during and after an earthquake of magnitude up to and including the safe shutdown earthquake. The control element assembly and buffer pin are seismic Category I equipment and will be qualified in accordance with the procedures outlined in CESSAR Section 3.9.1.2.

The mechanical properties of structural materials selected for the control rod system satisfy Appendix I of Section III of the ASME Code, or Part A of Section II of the Code, and also the staff position that the yield strength of cold worked austenitic stainless should not exceed 90,000 pounds per square inch.

Residual cold work in austenitic stainless steel is known to accelerate water corrosion. Expressing cold work in terms of increased yield strength, a yield strength of 90,000 pounds per square inch for Types 304 and 316 stainless steel corresponds to residual cold work greater than ten percent and less than twenty percent. The staff has selected a yield strength of 90,000 pounds per square inch as a conservative criterion for the use of cold worked austenitic stainless steel in light water reactor internals. This control imposed on the use of cold worked stainless steel will provide adequate protection during reactor operation from conditions which could lead to stress corrosion of the materials and loss of reactor internal structural integrity.

The proposed controls on the austenitic stainless steel in the system conform to the recommendations of Regulatory Guide 1.31 and Regulatory Guide 1.44. Fabrication and heat treatment practices that will be performed in accordance with these recommendations provide added assurance that stress corrosion cracking will not occur during the design life of the components. The compatibility of all materials to be used in the control rod system in contact with the reactor coolant satisfies the criteria for paragraphs NE-2160 and NB-3120 of Section III of the ASME Code. Both martensitic and precipitation-hardened stainless steels will be given tempering or aging treatments in accordance with staff positions. Conformance with the codes and Regulatory Guide recommendations mentioned above, and with the staff positions on allowable maximum yield strength of cold worked austenitic stainless steel and minimum tempering or aging temperatures of martensitic and precipitation-hardened stainless steels, constitutes an acceptable basis for meeting the requirements of the Commission's General Design Criterion 26.

We have concluded that the proposed mechanical design of the control rod system is acceptable.

4.2.4 Refueling Operations

The refueling procedures to be used at System 80 plants have been modified relative to those in use or to be used at previously licensed Combustion Engineering plants. The principal difference is that refueling is to be carried out with all control element assemblies removed. This change expedites the refueling process.

Following shutdown of the reactor, the primary system coolant temperature will be reduced to refueling values (<135 degrees Fahrenheit) and the refueling boron concentration will be established. Subsequent to removal of the head, an upper guide structure lifting rig will be locked to the upper guide structure. The control element assemblies will be individually withdrawn into the lifting rig and then this rig, the upper guide structure and the control element assemblies will be removed as a unit from the reactor vessel. The control element assemblies will not be inserted until refueling has been completed.

Prior to and during the refueling process, continuous flow of coolant through the core will be maintained, using the shutdown cooling system. This, together with frequent boron concentration measurements as required by the technical specifications, will ensure that the desired boron concentration will be maintained during the refueling process.

The safety of the refueling operations is not dependent on having the control element assemblies inserted in the core. With or without control element assemblies, the subcriticality of the system must be maintained using soluble boron. We require that the boron concentration be sufficient to maintain the core at least 5 percent subcritical, including uncertainties. As discussed in Section 15.4 of this report, we have evaluated the potential for boron dilution during refueling without control element assemblies and conclude that this method of refueling is acceptable.

4.3 Nuclear Design

The nuclear design of the System 80 reactor is in many respects similar to that of the Combustion Engineering 3390 thermal megawatt reactor design. The principal difference, a lower average linear heat generation rate (5.53 versus 6.9 kilowatts per foot), arises from the use of a 16 x 16 fuel rod array as opposed to the 14 x 14 design approved for use in earlier plants. Also, an increase in the number of fuel assemblies (217 to 241) results in a higher total reactor power with essentially no change in the power output per assembly.

The reactor will be operated at steady-state full power with the full-length control element assemblies virtually withdrawn. Limited insertion of full-length control element assemblies will be permitted to compensate for the effects of minor variations in moderator temperature and boron concentration. Part-length control element assemblies will be used to assist in the control of power distributions.

Soluble boron will be used to compensate for slow reactivity changes including those due to burnup and to changes in xenon concentration. The soluble boron control system (chemical and volume control system) will also provide the capability of bringing the reactor to at least 10 percent subcritical in the cold shutdown condition regardless of the positions of the control element assemblies.

Load changes will be accomplished using both the control element assemblies and the boron control system. Full-length control element assembly insertion at all powers will be controlled by the power-dependent insertion limits given in the technical specifications. These limits will ensure that (1) there is sufficient reactivity worth in the withdrawn control element assemblies to permit the rapid shutdown of the reactor with ample shutdown margin, and (2) that the worths of control element assemblies that might be ejected in the unlikely event of an ejected rod accident will be no worse than that assumed in the safety analysis. Alarms will be provided to ensure that these limits are not exceeded.

We have reviewed the calculated control element assembly worths and the uncertainties in these worths, and conclude that rapid shutdown capability exists assuming the most reactive control element assembly is stuck in the fully withdrawn position. In making this determination, we have considered the experimental information provided in the CESSAR to support the validity of the calculated control element assembly worths. We also conclude that sufficient allowance has been made in the calculated worths to account for calculational uncertainties.

Combustion Engineering specified a value of 2.28 as the design limit for the three-dimensional nuclear heat flux peaking factor (F_Q^N) including calculational uncertainties and the effects of densification. Combustion Engineering has calculated the power distributions expected during both steady-state and typical load-follow operations to show that the actual peaking factors can be maintained below the design values. An allowance for calculational uncertainty of 10 percent was assumed by Combustion Engineering in determining the expected peaking factors. We have reviewed comparisons between measured and calculated power distributions and found this uncertainty allowance acceptably conservative. Further, we concluded that the comparison between expected and design peaking factors demonstrated that the plant could be operated below the design value. Thus, a peak linear heat generation rate (PLHGR) of 13.3 kilowatts per foot[1/] was found to be acceptable for use in the accident analysis.

During our review, we required that changes be made to the loss-of-coolant accident analysis (see Section 6.3). The effect of these changes was, in part, a reduction in the allowable peak linear heat generation rate to a value of 12.1 kilowatts per foot. To comply with this limit, the total peaking factor, (F_Q^T), must be maintained at or below 2.15. The power distribution calculations performed by Combustion Engineering indicate that total peaking factor will not exceed 2.05 under steady-state, unrodded operating condition. This value is based on a product of the axial and radial peaking factors of 1.77, an engineering factor of 1.03, a nuclear uncertainty factor of 1.10 and an assumed densification penalty factor of 1.02. Thus we have concluded that the plant can be operated at steady-state, full power without exceeding the peaking factor limit. The margin between 2.05 and 2.15 is available to accommodate load following operations.

A reactor monitoring system, designated the core operating limit supervisory system, will be provided to monitor power distributions. This system will utilize the outputs of the incore detector system and the control element assembly position indicating system to continously monitor the power distributions to ensure that the operating limit on peak linear heat rate and margin to departure from nucleate boiling ratio are maintained. The system will also monitor azimuthal flux tilts and total power level and will generate alarms if any of these limits are exceeded. The core operating limit supervisory system functions will be executed in the core monitoring computer; a second plant computer is available to perform these functions if the core monitoring computer is unavailable. The use of incore detectors to monitor power distributions in System 80 plants is similar in concept to that which the staff has approved for use in previous Combustion Engineering plants. We have reviewed the design information submitted in CESSAR and are reviewing Topical Report CENPD-169-P "Assessment of the Accuracy of PWR Operation Limits as Determined by Core Operating Limit Supervisory System." This report provides a system description and the analysis of the errors associated with the core operating limit supervisory system processing. With the exception of the generic review matter discussed in Section 7.7, we find the core

[1/] $F_Q^T = F_Q^N \times F_Q^E = 2.28 \times 1.03 = 2.35$

where F_Q^E is the engineering factor

$PLHGR = F_Q^T \times 1.02 \times 5.53 = 13.3$ kilowatts per foot

Where 1.02 is the power measurement uncertainty factor and 5.53 kilowatts per foot is the average linear heat generation rate.

operating limit supervisory system acceptable for power distribution monitoring. We shall conduct a more detailed review of the core operating limit supervisory system functions during our review for a Final Design Approval with the intent of quantifying more precisely the accuracy of the processing methods used to produce a measured power distribution. Until that final review can be completed, we have accepted 10 percent as a conservative estimate of the error associated within the core operating limit supervisory system power distribution measurement.

Because of the increased effective diameter of the System 80 core (143 versus 136 inches for the 3390 thermal megawatt plant), the core will be more susceptible to radial and azimuthal oscillations. It is a design objective that the core be stable to both types of oscillations. Stability calculations based on the preliminary first cycle design show that this objective will be met. Combustion Engineering will provide results of final analyses of the first cycle at the Final Design Approval stage of our review, at which time the staff will make a final determination of the acceptability of these analyses.

4.4 Thermal and Hydraulic Design

The principal criterion for the thermal-hydraulic design of a reactor is avoidance of thermally induced fuel damage during normal steady-state operation and during anticipated operational occurrences. The CESSAR uses the following design limits to satisfy this criterion:

(1) The margin to departure from nucleate boiling will be chosen to provide a 95 percent probability with 95 percent confidence that departure from nucleate boiling will not occur on a fuel rod having the minimum departure from nucleate boiling ratio during normal operation and any anticipated operational occurrence. The preliminary design used a minimum allowable limit of 1.30 for the departure from nucleate boiling ratio.

(2) Operating conditions are selected to ensure hydraulic stability within the core, thereby preventing premature departure from nucleate boiling.

(3) The peak temperature of the fuel will be less than the melting point (2805 degrees centigrade unirradiated and reduced by 32 degrees centigrade per 10,000 megawatt days per metric ton of uranium during normal operation and anticipated operational occurrences).

The thermal and hydraulic design parameters for the reactor are listed and compared with those of San Onofre Units 2 and 3 in Table 4.1. The principal differences include increases in the allowable power, flow rate, inlet temperature, and the number of fuel assemblies. Present predictions of the hydraulic characteristics are based on model tests for other designs. Results of confirmatory flow model tests for the System 80 configuration will be submitted in the application for a Final Design Approval.

Within the core, the fuel assembly array will be changed from 14 x 14 to 16 x 16 fuel rods of reduced diameter and rod pitch, while the fuel assembly pitch will remain constant. This change together with the increase in the number of fuel assemblies, will increase the heat transfer surface area and reduce the peak heat flux and linear heat rate, thereby increasing the thermal margin for a given core power density. Further benefits in thermal margin will be obtained by utilizing V-tab mixing grids in the fuel bundles.

The margin to departure from nucleate boiling at any point in the core is expressed in terms of the departure from nucleate boiling ratio. The departure from nucleate boiling ratio is defined as the ratio of the heat flux required to produce departure from nucleate boiling at the calculated local coolant conditions to the actual local heat flux. The departure from nucleate boiling correlation used for the design of this core is the Westinghouse Electric Corporation W-3 correlation modified by a constant multiplier of 1.03. Data in the open literature[1/] indicate that 1.03 is a conservative multiplier for use with tab type grids. Combustion Engineering has committed to provide a topical report describing the ongoing departure from nucleate boiling tests. The report is scheduled for issuance by June 1976. Results for full length uniform and non-uniform axial heat flux departure from nucleate boiling tests and the statistical data analysis for 16 x 16 fuel geometry and V-tab spacer grids that acceptably demonstrates compliance with design limit (1), above, will be submitted in the application for a Final Design Approval.

The reactor core will be designed using the TORC code, an open core analytical method based on the COBRA-III-C model. The TORC code solves the conservation equations for mass, axial and lateral momentum, and energy for a collection of parallel flow channels that are hydraulically open to each other. The principal revisions to COBRA-III-C which transpose COBRA-III-C to TORC are summarized in the CESSAR.

Combustion Engineering has submitted a topical report describing TORC for our review. This report includes a description of data used to verify the TORC code on a subchannel basis. Combustion Engineering has also committed to provide an additional topical report that will use existing or soon-to-be obtained reactor data to verify the TORC code on a core-wide basis. This report is scheduled for issuance by June 1976. We plan to review the topical reports describing the TORC code for adequacy, prior to the submittal of the CESSAR Final Safety Analysis Report.

Another parameter that influences the thermal-hydraulic design of the core is rod-to-rod bowing within fuel assemblies. The bowing effect is being reviewed generically, and if rod-to-rod bowing proves to be a significant problem for the 16 x 16 fuel design, penalties may be imposed at the operating license stage of our review.

On the basis of our review of the design parameters and limits, the predicted hydraulic characteristics and Combustion Engineering commitments to (1) perform confirmatory

1/ E. R. Rosal, et al., "High Pressure Rod Bundle DNB Data with Axially Non-Uniform Heat Flux," Nuclear Engineering and Design, 31, 1974, pp 1-20.

flow tests, (2) verify the departure from nucleate boiling correlation, (3) demonstrate compliance with the 95/95 departure from nucleate boiling criterion, and (4) verify the TORC code as discussed above, we conclude that the proposed thermal-hydraulic design, including the design differences between the System 80 design and previous Combustion Engineering designs, is acceptable.

We will review the Combustion Engineering topical reports concerning departure from nucleate boiling and the TORC code and the results of the flow tests when they are submitted and we will require the thermal-hydraulic analysis to be substantiated by this additional information in the application for Final Design Approval.

5.0 REACTOR COOLANT SYSTEM

5.1 Summary

The proposed reactor coolant system will circulate water in a closed cycle, removing heat from the reactor core and internals and transferring it to the steam generators. It will include a reactor vessel and two coolant loops, each loop having a steam generator. Each coolant loop will consist of a hot leg between the reactor vessel outlet and the steam generator inlet and two cold legs from the steam generator outlets to the reactor vessel inlet. Each cold leg will contain a reactor coolant pump. Coolant system pressure will be maintained by a pressurizer that will be connected to one of the two hot legs. All system components will be located in the containment building.

The System 80 reactor coolant system design presented in the CESSAR is similar to that reviewed and approved for San Onofre Units 2 and 3; however, it includes the following significant differences:

(1) Steam generators that incorporate internal economizers and operate at higher steam temperatures.

(2) A reactor drain tank to receive and condense the pressurizer relief discharge and elimination of a separate pressurizer relief tank provided for previous plants.

(3) The addition of instrument nozzles to the bottom head of the reactor vessel and the provisions of guide tubes inside the lower plenum of the reactor vessel to accommodate bottom entry, moveable in-core neutron flux detectors.

(4) A larger reactor vessel and higher coolant flow rate required by the increased number of fuel elements and higher power level.

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary Components

The design loading combinations specified for American Society of Mechanical Engineers (ASME) Code Class 1 reactor coolant pressure boundary components have been appropriately categorized with respect to the plant conditions identified as normal, upset, emergency or faulted. The design limits proposed by Combustion Engineering for these plant conditions are consistent with the recommendations of Regulatory Guide 1.48. Use of these recommendations for the design of reactor coolant pressure boundary components will provide reasonable assurance that if an earthquake should occur at the site or if upset, emergency or faulted conditions should develop, the resulting combined stresses imposed on the system components will not exceed the allowable design stresses and strain limits for the materials of construction. Limiting the stresses and strains under such loading combinations provides a basis for the design of the system components for the most adverse loadings postulated to occur during the service lifetime without loss of the system's structural integrity. The design load combinations and associated stress and deformation limits specified for ASME Code Class 1 components constitute an acceptable basis for design in satisfying the related requirements of the Commission's General Design Criteria 1, 2 and 4.

5.2.1.1 Compliance With 10 CFR Part 50, Section 50.55a

We have reviewed the information provided in the CESSAR and conclude that pressure-retaining components of the reactor coolant pressure boundary as defined by the rules of 10 CFR Part 50, Section 50.55a, have been properly identified in CESSAR Table 5.2-1 and classified as ASME Section III, Code Class 1 components. Combustion Engineering states that reactor coolant pressure boundary components will be constructed in accordance with the requirements of the applicable codes and addenda as specified by the rules of 10 CFR Part 50, Section 50.55a. In conformance with these requirements, the code edition and the applicable addenda for each ASME Section III, Code Class 1 component will be specified in applications referencing or incorporating the CESSAR System 80 design.

We conclude that construction of the components of the reactor coolant pressure boundary in conformance with the ASME code and the Commission's regulations provides adequate assurance that component quality will be commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

5.2.1.2 Applicable Code Cases

We have reviewed the ASME code cases identified in Table 5.2-6 of the CESSAR. These requirements will be applied in the construction of pressure-retaining ASME Section III, Code Class 1 components of the reactor coolant pressure boundary (Quality Group Classification A). The code cases specified in Table 5.2-6 are in accordance with those code cases in Regulatory Guides 1.84, "Code Case Acceptability ASME Section III Design and Fabrication," and 1.85, "Code Case Acceptability ASME Section III Materials."

We conclude that compliance with the requirements of these code cases, in conformance with the Commission's regulations, will result in a component quality level that is commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

5.2.2 Overpressurization Protection

The reactor coolant system design relies upon the combined action of the pressurizer safety valves, the steam system safety valves and the reactor protection system for overpressurization protection. The System 80 design scope includes the pressurizer safety valves and the reactor protection system. The design of the steam and feedwater system piping and valves, including the relief and safety valves to protect the steam generator shell side against overpressurization, will be provided by users referencing the CESSAR. The relief requirements for the shell side of the steam generator that are necessary to protect both the reactor coolant system and the shell side of the steam generator are specified in the CESSAR as design interface requirements that must be met by the user. This design interface requires a total shell side relief flow rate of 19 x 10^6 pounds per hour, minimum, at valve set pressure.

The overpressurization protection will be designed to limit the reactor coolant system primary and secondary side pressure to less than 110 percent of their respective design pressures of 2500 and 1512 pounds per square inch, absolute, following a one hundred percent loss of turbine generator load without a simultaneous reactor trip, i.e., the reactor is assumed not to trip until it receives a high reactor coolant system pressure signal. Combustion Engineering has calculated that the maximum primary side pressure will be approximately 104 percent of the design pressure during this transient.

The criteria used in developing the design and mounting of ASME Class 1 safety and relief valves for the System 80 design provides adequate assurance that, under discharging conditions, the resulting stresses will not exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design of the system components to withstand these loads without loss of structural integrity and impairment of the overpressure protection function.

We conclude that the criteria used for the design and installation of ASME Class 1 overpressure relief devices are consistent with Regulatory Guide 1.67, and constitute an acceptable design basis in meeting the applicable requirements of the Commission's General Design Criteria 1, 2, 4, 14 and 15.

5.2.3 Reactor Coolant Pressure Boundary Materials

The construction materials for components of the reactor coolant pressure boundary will be in conformance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, including addenda and code cases appropriate to comply with Appendix A of 10 CFR Part 50. The residual elements (copper, phosphorous, sulfur and vanadium) in the ferritic material specified for the reactor vessel will be limited by Combustion Engineering in order to reduce the effect of irradiation on the fracture toughness of the materials in the reactor vessel beltline. The predicted shift in the referenced nil-ductility temperature is in agreement with the recommendations of Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

Austenitic stainless steels will be used for construction of pressure-retaining components in the reactor coolant pressure boundary. Unstabilized austenitic Types 304 and 316 stainless steel will normally be used. Because these compositions are susceptible to stress corrosion cracking when exposed to certain environmental conditions, process controls will be exercised during all stages of component manufacturing and reactor construction to avoid sensitization of the materials that could lead to stress corrosion cracking.

The controls imposed upon components constructed of austenitic stainless steel used in the reactor coolant pressure boundary conform with the recommendations of Regulatory Guides 1.31, "Control of Stainless Steel Welding," and 1.44, "Control of the Use of Sensitized Stainless Steel." Material selection, fabrication practices, and examination and protection procedures performed in accordance with these recommendations provide reasonable assurance that the austenitic stainless steel in the reactor coolant pressure boundary will be free from hot cracking (microfissures) and in a metallurgical condition which precludes susceptibility to stress corrosion cracking during service. Conformance with Regulatory Guides 1.31 and 1.44 constitutes an acceptable basis for meeting the requirements of General Design Criteria 1 and 14.

The materials of construction of the reactor coolant pressure boundary that will be exposed to the reactor coolant have been reviewed and all of the materials are compatible with the expected environment. General corrosion of all materials, except unclad carbon and low alloy steel, will be negligible. Conservative corrosion allowances have been provided for these materials in accordance with the requirements of Section III of the ASME Code.

The reactor coolant system water chemistry is selected to minimize corrosion. Periodic analysis of the chemical composition will be performed to verify that the coolant water quality conforms to the specification. The chemical and volume control system will provide means for adding chemicals to the coolant to scavenge oxygen and to control the pH. Hydrazine and hydrogen will be used to scavenge oxygen, and lithium hydroxide will be used to control pH. The controls imposed on reactor coolant and auxiliary systems fluid chemistry are in conformance with the recommendations of Regulatory Guide 1.44. This conformance provides reasonable assurance that the reactor coolant pressure boundary components will be adequately protected during operation from conditions that could lead to stress corrosion of the materials and loss of structural integrity of a component.

The monitoring instrumentation of the reactor coolant water chemistry will provide acceptable capability to detect changes on a timely basis to effect corrective actions before stress corrosion attack occurs at an unacceptable level. The use of materials of proven performance and in conformance with the recommendations of Regulatory Guide 1.44 constitutes an acceptable basis for satisfying the requirements of the Commission's General Design Criteria 14 and 31.

712 245

The controls to be imposed on welding preheat temperatures will be in conformance with the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel." These controls provide reasonable assurance 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 retained in the weldment.

The codes, standards and specifications proposed in CESSAR Section 5.5.2 as the basis for selecting and fabricating the materials to be used in the Class 1 components of the steam generators are acceptable to the staff. Conformance with these proposed codes constitutes an acceptable basis for meeting the requirements of the Commission's General Design Criteria 14, 15, 31 and 32.

5.2.4 Fracture Toughness

We have reviewed the materials selection, toughness requirements, and materials testing proposed in the CESSAR to provide assurance that the ferritic materials used for pressure retaining components of the reactor coolant pressure boundary will have adequate toughness under test, normal operation, and transient conditions. The ferritic materials are specified to meet the toughness requirements of the ASME Code Section III, including Summer 1972 Addenda. In addition, materials for the reactor vessel are specified to meet the additional test requirements and acceptance criteria of Appendix G of 10 CFR Part 50.

The fracture toughness tests and procedures required by Section III of the ASME Code as augmented by Appendix G of 10 CFR Part 50, for the reactor vessel and all pressure retaining components of the reactor coolant pressure boundary, provide reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating fracture can be established during operating, testing, maintenance, and postulated accident conditions. Conformance with the code provisions and Commission regulations constitutes an acceptable basis for satisfying the requirements of the Commission's General Design Criterion 31.

The reactor will be operated in accordance with the ASME Code, Section III, including Summer 1972 Addenda, and Appendix G of 10 CFR Part 50. This will minimize the possibility of failure due to a rapidly propagating crack. Additional conservatism in the pressure-temperature limits used for heatup, cooldown, testing, and core operation will be provided because the pressure-temperature limits will be determined assuming that the beltline region of the reactor vessel has already been irradiated.

The use of Appendix G of the code as a guide in establishing safe operating limits using results of the fracture toughness tests performed in accordance with the code and the Commission's regulations, will ensure adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Conformance with these code provisions and the Commission's regulations constitutes an acceptable basis for satisfying the requirements of the Commission's General Design Criterion 31.

The toughness properties of the reactor vessel beltline material will be monitored throughout service life with a material surveillance program that will meet the requirements of the American Society for Testing Materials Standard E-184-73. This program also complies with Appendix H of 10 CFR Part 50, except that specimen holders will be attached to the vessel cladding. Combustion Engineering has submitted a Topical Report, CENPD-155P, "CE Procedure for Design, Fabrication, Installation and Inspection of Surveillance Specimen Holder Assemblies," dated September 1974. We have evaluated this report and concluded, based on our evaluation as presented in our letter to Combustion Engineering, dated May 15, 1975, that the procedures for design, fabrication, installation and inspection of surveillance specimen holder assemblies described in this report are acceptable. On the basis of the information provided in CENPD-155P, we conclude that the method of attaching capsule holders to the vessel clad is acceptable and results in no degradation of the vessel base material. Requests for exemptions from this requirement of Appendix H of 10 CFR Part 50 will be considered on individual plant applications referencing CESSAR.

Changes in the fracture toughness of material in the reactor vessel beltline caused by exposure to neutron radiation will be assessed properly, and adequate safety margins against the possibility of vessel failure will be provided if the material requirements of the above specifications and regulations are met. Conformance with these specifications and regulations will ensure that the surveillance program constitutes an acceptable basis for monitoring radiation induced changes in the fracture toughness of the reactor vessel material, and will satisfy the requirements of the Commission's General Design Criterion 31.

The use of controlled composition material for the reactor vessel beltline will minimize the possibility that radiation will cause serious degradation of the toughness properties. In addition, Combustion Engineering has stated that if the results of tests indicate that the toughness is not adequate, the reactor vessel can be annealed to restore the toughness to acceptable levels. We concur with this statement.

5.2.5 Austenitic Stainless Steel

The controls to be imposed upon components constructed on austenitic stainless steel used in the reactor coolant pressure boundary conform with the recommendations of Regulatory Guides 1.31, and 1.44. Materials selection, fabrication practices, examination procedures, and protection procedures performed in accordance with these recommendations provide reasonable assurance that the austenitic stainless steel in the reactor coolant pressure boundary will be free from hot cracking (microfissures) and in a metallurgical condition which precludes susceptibility to stress corrosion cracking during service. Conformance with these Regulatory Guides constitutes an acceptable basis for meeting the requirements of the Commission's General Design Criteria 1 and 14.

5.2.6 Pump Flywheel

The probability of loss of pump flywheel integrity will be minimized by the use of suitable material, adequate design, and inservice inspection. Combustion Engineering has stated that the reactor coolant pump flywheel will conform with the recommendations of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity." The use of suitable material, and adequate design and inservice inspection for the flywheels of reactor coolant pump motors as recommended in the CESSAR provides reasonable assurance that (1) the structural integrity of the flywheels will be adequate to withstand the forces imposed in the event of a pump design overspeed transient without loss of function, and (2) the integrity of the flywheels will be verified periodically, in service, to assure that the soundness of the flywheel material is maintained at a level adequate to preclude failure. Conformance with the recommendations of Regulatory Guide 1.14 constitutes an acceptable basis for satisfying the requirements of the Commission's General Design Criterion 4.

5.2.7 Leakage Detection System

Coolant leakage within the primary containment may be an indication of a small through-wall flaw in the reactor coolant pressure boundary. Although the design of the leakage detection system will be provided in the balance of plant design, the CESSAR specifies interface requirements for such a system. The CESSAR specifies that the design of the leakage detection system proposed for detection of leakage to the containment will include diverse leak detection methods, will have sufficient sensitivity to measure small leaks, will identify the leakage source to the extent practical, and will be provided with suitable control room alarms and readouts. These interface requirements assure that the leakage detection systems will conform to the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." Thus, reasonable assurance is provided that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions. Conformance with the recommendations of Regulatory Guide 1.45 constitutes an acceptable basis for satisfying the requirements of the Commission's General Design Criterion 30.

5.2.8 Inservice Inspection Program

To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be inspected periodically. Combustion Engineering has stated that the design of the reactor coolant system will incorporate provisions for access for inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and that suitable equipment will be developed to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel. The conduct of periodic inspections and hydrostatic testing of pressure retaining components in the reactor coolant pressure boundary in accordance with the requirements of Section XI of the ASME Code provides reasonable assurance that evidence of structural degradation or loss of leaktight-integrity occurring during service will be detected in time to permit corrective action before the safety function of a component is compromised. Conformance with the inservice inspections required by this code constitutes an acceptable basis for satisfying the requirements of the Commission's General Design Criterion 32.

To ensure that no deleterious defects develop during service in ASME Code Class 2 system components, selected welds and weld heat-affected zones will be inspected prior to reactor startup and periodically throughout the life of the plant. Code Class 2 systems and Code Class 3 systems will receive visual inspections while the systems are pressurized in order to detect leakage, signs of mechanical or structural distress, and corrosion. Examples of Code Class 2 systems are residual heat removal systems, portions of chemical and volume control systems, and those engineered safety features not part of Code Class 1 systems. Examples of Code Class 3 systems are component cooling water systems, and portions of the radwaste systems. Combustion Engineering has stated that the Code Class 2 systems will meet the requirements of the ASME Code, Section XI. The requirements for Code Class 2 systems and Code Class 3 systems will be in conformance with the recommendations of Regulatory Guide 1.51, "Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components." Conformance with the inservice inspections required by this ASME code and Regulatory Guide 1.51 constitutes an acceptable basis for satisfying the Commission's General Design Criteria 36, 39, 42, and 45.

5.2.9 Loose Parts Monitor

Occasionally, miscellaneous items such as nuts, bolts, and other small items have become loose parts within reactor coolant systems. In addition to causing operational inconvenience, such loose parts can damage other components within the system or be an indication of undue wear or vibration. For the past few years we have required each applicant to initiate a program, or to participate in an ongoing program, the objective of which was the development of a functional, loose parts monitoring system within a reasonable period of time. Recently, prototype loose parts monitoring systems have been developed and are presently in operation or are being installed at several plants. The CESSAR has imposed an interface requirement (Section 4.2.4.J) stating that a loose parts monitoring system shall be provided by the balance of plant designs that reference the CESSAR. We have concluded that this is an acceptable basis for Preliminary Design Approval.

5.3 Thermal Hydraulic System Design

The thermal and hydraulic design bases for the reactor coolant system are discussed in Section 4.4 of this report.

5.4 Reactor Vessel and Appurtenances

We have reviewed all factors contributing to the structural integrity of the reactor vessel, and conclude that there are no special considerations that make it necessary to consider potential vessel failure in evaluating the consequences of design basis accidents. The bases for our conclusion are that the design, material, fabrication, inspection, and quality assurance requirements for the reactor vessel will conform to the ASME Code, Section III, including the 1972 Summer Addenda. Also, operating limitations on temperature and pressure will be established for the plant in accordance with Appendix G, "Protection Against Nonductile Failure," of the 1972 Summer Addenda of the ASME Code, Section III, and Appendix G of 10 CFR Part 50.

The integrity of the reactor vessel is assured because the vessel:

(1) Will be designed and fabricated to the high standards of quality required by the ASME Code, Section III and pertinent code cases.

(2) Will be made from materials of controlled and demonstrated high quality.

(3) Will be inspected and tested to provide substantial assurance that the vessel will not fail because of material or fabrication deficiencies.

(4) Will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation or during most upsets in operation, and that the vessel will not fail under the conditions of any of the postulated accidents.

(5) Will be subjected to monitoring and periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under the service conditions.

(6) May be annealed to restore the material toughness properties if this becomes necessary.

The reactor vessel closure studs will be designed and initially inspected in conformance with the recommendations of Regulatory Guide 1.65, "Materials and Inspection for Reactor Vessel Closure Studs." We therefore have concluded that the design of the studs is acceptable. Inservice inspection requirements for reactor vessel studs are not within the CESSAR scope, but will be provided by the utility-user.

5.5 Components and Subsystem Design

5.5.1 Reactor Coolant Pumps

The reactor coolant pumps will be sized to deliver flow at rates which equal or exceed the required flow rates under normal and transient operating core conditions. A limit on low reactor coolant pump flow rate (111,400 gallons per minute) has been established to assure that specified fuel design limits will not be exceeded.

The four reactor coolant pumps will be vertical, single bottom suction, horizontal discharge, motor-driven centrifugal pumps. The pump impeller will be keyed and locked to its shaft. Pump shaft alignment will be maintained by a water lubricated bearing within the pump and by radial and thrust bearings. The pump and motor shafts will be connected by a coupling. Each motor will be provided with an antireverse rotation device.

712 250

Pump rotational kinetic energy (32,000,000 foot pounds) is to be provided by a flywheel, in conjunction with the impeller and motor assembly, to provide flow during coastdown in the event of a loss of pump power. Sufficient kinetic energy will be provided by the flywheel, impeller and motor assembly when operating at normal speed to provide adequate flow during a coastdown event following a loss of pump power while operating at the normal rotational speed associated with 60 Hertz pump power. An interface requirement in the CESSAR assures that underfrequency decay rate events cannot result in a more rapid decrease in reactor coolant flow rate than has been assumed in the CESSAR System 80 accident analysis.

5.5.2 Steam Generator

The proposed steam generators are vertical shell, U-tube evaporators with an integral economizer and integral moisture separating equipment. Hot reactor coolant will flow through the U-tubes, heating and evaporating the feedwater on the shell side to produce steam. The tube and tube sheet boundary will be designed to prevent the transfer of activity generated within the reactor core to the steam system. Since the steam generators will provide a heat sink for the reactor coolant system, they will be at a higher elevation than the reactor core. The elevation difference will create a natural circulation capability sufficient to remove core decay heat following coastdown of all reactor coolant pumps.

Criteria 14, 15 and 31 of the General Design Criteria require that the reactor coolant pressure boundary have an extremely low probability of abnormal leakage and be designed with sufficient margin to assure that design conditions will not be exceeded during normal operation and anticipated operational occurrences, and that the probability of rapidly propagating failure of the reactor coolant pressure boundary will be minimized. The steam generator forms an important part of the boundary.

We have reviewed the selection of steam generator materials and the controls which will be exercised during the fabrication of these components. The steam generators will be fabricated as ASME Boiler and Pressure Vessel Code Class 1 and 2 components. The mechanical properties for the materials selected for the steam generators will meet ASME Code requirements as stated in Appendix I of Section III and Parts A, B and C of Section II of the Code. Welding procedures and fabrication processes will be qualified in accordance with the requirements of Sections III and IX of the ASME Code. The Class 1 components of the steam generator will meet the fracture toughness requirements of applicable Code Addenda, including Article NB-2300, Section III, ASME Code, Summer 1972 Addenda and Appendix G, Paragraph G-2000. Class 2 steam generator materials will meet the fracture toughness requirements of applicable Code Addenda, including Paragraphs NC-2310 and NC-2320, Section III, ASME Code, Summer 1972 Addenda.

The procedures for weld-depositing corrosion-resistant cladding on the tube sheet will be qualified according to the requirements of Article Q-12 of Section IX of the ASME Code. The Inconel 600 tubes will be expanded for the full depth of the tube sheet to avoid the presence of a deep crevice between the tube and tube sheet pursuant to the recommendations of Branch Technical Position MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators." The welds between the tubes and tube sheet will meet the requirements of Sections III and IX of the ASME Code.

Onsite cleaning and cleanliness control will be in accordance with the intent of the recommendation of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," as stated in ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants."

Conformance with the above stated applicable codes, standards, staff positions, and Regulatory Guides constitutes an acceptable basis for meeting the requirements of General Design Criteria 14, 15 and 31.

The System 80 generators differ from steam generators provided by Combustion Engineering for previously reviewed and approved plants in that the System 80 steam generators have integral economizers. Instead of introducing feedwater through a sparger ring to mix with the recirculating flow in the downcomer channel, feedwater will be introduced into a separate, but integral section of the steam generator. A semi-cylindrical section of the tube bundle, at the exit end of the U-tubes, will be separated from the remainder of the tube bundle by vertical divider plates and horizontal baffle pla. s. Feedwater will be directly introduced into this section and pre-heated before entering the tube bundle region. Section 1.5.5 of the CESSAR describes a development program that Combustion Engineering is conducting to confirm the steam generator structural integrity during thermal transients, and main steam line and feedwater line break accidents. Section 1.5.5 of the CESSAR also states that the results of this program will be submitted in a topical report for the Commission's review by December 31, 1976. The economizer box will be designed to withstand the consequences of postulated feedline rupture. A detailed stress analysis showing conformance with this requirement will be submitted in the application for Final Design Approval.

On the basis of our review of the proposed design, and Combustion Engineering's commitment to confirm the adequacy of the design by a development program and detailed analyses that will be submitted in the application for Final Design Appr val, we conclude that the proposed steam generator design is acceptable

The staff has under consideration appropriate monitoring of secondary water chemistry and in service inspection programs to further enhance steam generator tube integrity. Upon completion of our review we will consider appropriate recommendations or requirements for use in connection with e CESSAR design.

5.5.3 Reactor Drain Tank

Previous nuclear steam supply systems designed by Combustion Engineering used both pressurizer relief tanks and reactor drain tanks. The System 80 design uses the reactor drain tank to receive and condense the design discharge from the pressurizer safety valves as well as to receive and store drainage from reactor coolant system leakoff locations. The reactor drain tank will have a total volume of 376 cubic feet and will be required to have a minimum water volume of 180 cubic feet for steam quenching due to pressurizer relief valve discharge. The tank is designed to condense

1500 pounds of steam with this minimum water volume and at the same time provide an additional space for 1000 gallons of water drainage from the reactor coolant system. A low pressure nitrogen gas blanket will be maintained to exclude oxygen.

The condensing capacity of the drain tank will be more than the combined total steam discharge due to control element assembly withdrawal and loss of load transients.

We have reviewed Combustion Engineering's proposed design of the reactor drain tank and conclude it is sufficiently large to condense the total steam discharge due to any anticipated operational occurrence without rupturing the tank's rupture disc. We, therefore, conclude that the drain tank design is acceptable.

5.5.4 Safety Valves

The spring loaded primary safety valvas will be designed to protect the reactor coolant system as required by Section III of the ASME Boiler and Pressure Vessel Code. The design of the safety valves will be based upon a loss-of-load from maximum expected power level.

In sizing the safety valves, Combustion Engineering has assumed that a loss-of-load will not trip the reactor immediately, but that a delayed reactor trip will occur due to a high pressurizer pressure signal. No credit has been taken for the action of the pressurizer spray, the letdown flow, heat transfer to pressurizer walls, or the turbine bypass system, but credit was taken for action of the steam system safety valves and the reactor trip caused by the high pressurizer pressure. The calculated primary safety valve flow rate is less than the total rated capacities of the safety valves (40,500 versus 1,540,000 pounds per hour). The large difference in the flow rate is due to Combustion Engineering's sizing procedure which sizes the safety valves based on turbine trip without reactor trip. We find the resulting 1,540,000 pounds per hour total relieving capacity of the safety valves sufficiently conservative and acceptable.

Plants such as Calvert Cliffs Units 1 and 2 (Docket Nos. 50-317 and 50-318) utilize power operated relief valves to avoid opening of the spring loaded safety valves following reactor trip. The CESSAR design does not have power operated relief valves. Consequently, following a reactor trip, primary system pressure in System 80 plants might reach the primary safety valve set pressure. However, this will not cause any undue risk to the primary coolant system as long as the total relieving capacity for the safety valves is adequate to handle the overpressure transient.

The steam system safety valves are not in the CESSAR System 80 design scope, but will be designed by the balance of plant designer or utility-user who will be required to meet the System 80 design interface requirements for safety valve relief pressure and flow rate. We will review this aspect of the safety valves in applications which reference or incorporate the CESSAR System 80 design.

712 253

5.5.5 Residual Heat Removal

Residual heat removal will be accomplished by use of the shutdown cooling system. The shutdown cooling system will be used in conjunction with the main steam and the main and auxiliary feedwater systems to reduce the temperature of the reactor coolant system, in post-shutdown periods, from normal operating temperature to the refueling temperature. The initial phase of the cooldown will be accomplished by heat rejection from the steam generators to the turbine condenser or the atmosphere. After the reactor coolant temperature and pressure have been reduced to approximately 350 degrees Fahrenheit and 400 pounds per square inch, absolute, the shutdown cooling system will be put into operation to reduce the reactor coolant temperature to the refueling temperature and to maintain this temperature during refueling.

The shutdown cooling system will consist of two redundant loops, each containing a shutdown cooling heat exchanger, and a low-pressure safety injection pump. Three normally closed motor operated valves in each shutdown cooling suction line, in addition to the manual valves, will provide for isolation of the shutdown cooling system from the reactor coolant system.

During shutdown cooling, a portion of the reactor coolant will flow through the shutdown cooling nozzles located on the hot leg pipes, and will be circulated through the shutdown cooling heat exchangers by the low-pressure safety injection pumps and returned to the reactor coolant system through the four low-pressure safety injection lines. To increase the rate of cooldown during the latter stages, the containment spray pumps may be used to assist the low-pressure safety injection pumps in circulating the reactor coolant through the shutdown cooling system, provided the reactor coolant temperature is less than 200 degrees Fahrenheit.

The cooldown rate will normally be controlled by adjusting flow through the heat exchangers with throttle valves on the discharge of each heat exchanger. The flow controller will maintain a constant total shutdown cooling flow to the core by adjusting the heat exchanger bypass flow to compensate for changes through the heat exchangers.

We have reviewed the proposed System 80 shutdown cooling system design find that it meets the following requirements: (1) a single mechanical failure will not incapacitate the system; (2) an appropriate number and arrangement of isolation valves are provided to prevent overpressurization of the system; and (3) interface requirem are provided in the CESSAR to ensure that normal plant cooldown operation can be accomplished from the control room while experiencing the most limiting single failure. We will require the balance of plant designer to supplement the interface requirements specified in Section 5.1.4.I, Item 7, in Amendment 41 of the CESSAR as follows: The atmospheric dump valves associated with a steam generator shall be capable of holding the plant at hot standby dissipating core decay and reactor coolant pump heat, and allowing controlled cooldown from hot standby to shutdown cooling system initiation conditions. This will allow a controlled plant cooldown in the event of a steam line

712 254

break (which empties one steam generator) concurrent with the single active failure in the remaining two atmospheric dump valves. To accomplish the above, each atmospheric dump valve associated with a single steam generator shall have a saturated steam capacity of not less than 950,000 pounds per hour at 1000 pounds per square inch, absolute, assuming a single failure. This interface was contained in Amendment 36 to the CESSAR, and was found acceptable by the Commission staff, but was subsequently modified in the CESSAR. Based on the above, we conclude that the residual heat removal system described in the CESSAR is acceptable.

5.6 Design Interface Requirements

The CESSAR specifies design interface requirements that must be met by the balance of plant designer or utility-user in order to assure that the assumptions concerning the balance of plant, that were made by Combustion Engineering in its design and evaluation of the CESSAR reactor coolant and shutdown cooling systems, are valid and that the systems will meet their specified functional design requirements.

We have reviewed these interface requirements and conclude that they adequately specify balance of plant design requirements related to the reactor coolant and shutdown cooling system; therefore, subject to the modification of the atmospheric steam dump interface, we conclude that they are acceptable.

6.0 ENGINEERED SAFETY FEATURES

6.1 General

Engineered safety features is the designation given to those systems which are provided for the protection of the public and station personnel against the postulated release to the environment of radioactive products from the nuclear plant, particularly as the result of a loss-of-coolant accident. This section discusses the emergency core cooling system, the containment isolation system, and the mass and energy releases due to loss-of-coolant and steam line break accidents that must be used by the balance of plant designer in establishing the design criteria for the containment structure.

Certain of these systems will have functions for normal plant operation as well as serving as engineered safety features. Systems and components designated as engineered safety features will be designed to be capable of assuring safe shutdown of the reactor under the adverse conditions of the various postulated design basis accidents described in Section 15 of this report. They will be designed, therefore, to seismic Category I standards and must function even with complete loss of offsite power. Components and systems will be provided in sufficient redundancy so that a single failure of any component or system will not result in the loss of the capability to achieve safe shutdown of the reactor. The instrumentation systems and emergency power systems will be designed to the same seismic and redundancy requirements as the systems they serve. These systems are described in Sections 7 and 8 of this report.

6.2 Containment Systems

The containment systems for a nuclear generating station utilizing the CESSAR System 80 design will include a reactor containment structure, containment heat removal system, containment isolation system, containment combustible gas control system and containment leakage testing provisions. Except for a portion of the containment isolation system, these systems are not within the scope of the CESSAR System 80 design; however, Combustion Engineering has provided design interfaces required for proper mating of the CESSAR systems and those provided with the balance of plant design. We reviewed the interface information contained in the CESSAR and concluded that it provides an acceptable basis with respect to design requirements imposed on the balance of plant for the containment system.

6.2.1 Containment Functional Design

The containment building design will be provided by the utility-user in its construction permit application which incorporates or references the CESSAR. The containment structure will provide a low leakage barrier and will enclose the nuclear steam supply system, which includes the reactor, steam generators, reactor coolant pumps and pressurizer and certain components of the plant's engineered safety feature and auxiliary systems.

The CESSAR provides the mass and energy releases that would result from a spectrum of loss-of-coolant accident break sizes in the reactor coolant system, as well as various ruptures in the steam system piping. These data maximize the energy release to the containment as discussed below, and establish the design requirements (pressure and temperature) for the containment building for use by the balance of plant designer. These assumptions are, therefore, different from those used in the containment pressure calculations for emergency core cooling system evaluations. That is, the emergency core cooling system calculations are made conservative by minimizing containment pressure as discussed in Section 6.2.3 of this report, whereas the analysis of containment functional design is made conservative by maximizing pressure. Mass and energy releases are provided for a spectrum of primary system break sizes in the cold leg piping at the suction and discharge sides of the primary coolant pump, as well as a spectrum of hot leg break sizes (Tables 6.2.1-7 to 6.2.1-21 inclusive in the CESSAR). A spectrum of secondary system ruptures, as a function of power level, are also provided (Tables 6.2.1-22 to 6.2.1-30 inclusive in the CESSAR). Based on our evaluation of containment response for reactor plants similar to the CESSAR System 80 design, we believe that the break sizes provided will be sufficient to establish the design basis accident for the containment. However, if the containment pressure analysis performed by the balance of plant designer indicates that the most severe break size is outside the spectrum for which energy releases have been provided within the CESSAR, we will require that additional break sizes be analyzed on an individual plant basis.

The mass and energy released to the containment from a loss-of-coolant accident is considered in terms of blowdown, refill, reflood, and post-reflood phases. The blowdown phase of the accident is the time immediately following the occurrence of the postulated break during which most of the energy contained in the reactor system, including the primary coolant, metal and core stored energy is released to the containment. The refill phase is that time during which the lower reactor vessel plenum is refilled to the bottom of the core by the emergency core cooling system. The reflood phase is that time during which the core is reflooded to a 10-foot elevation and, for cold leg breaks, the time period during which most of the secondary energy is removed from the steam generators. The remaining energy in the secondary system, along with decay heat from the reactor core, is released to the containment during the post-reflood period. For hot leg breaks the broken piping provides a direct path for fluid from the core to travel directly into the containment without passing through the steam generators. Therefore the secondary energy will be removed at a much slower rate.

712 257

The CEFLASH-4 computer code was used by Combustion Engineering to determine the mass and energy addition rates to the containment during the blowdown phase of the accident; this code was found to be acceptable to the staff as indicated in our letter to Combustion Engineering dated June 13, 1975. To obtain a conservatively high energy release to the containment during the blowdown phase, Combustion Engineering has assumed that the core would remain in nucleate boiling for an extended period of time, so that the energy release rate from the core would be maximized. Under this assumption, the core transfers more heat to the containment than would be predicted by a calculation suitable for core heatup and emergency core cooling performance evaluation. This additional energy release from the core will increase the calculated containment pressure and therefore assures a margin of conservatism in the analysis.

The time delay for the lower plenum to be refilled to the core bottom has not been considered by Combustión Engineering for containment analysis. Combustion Engineering has conservatively assumed that the bottom of the core is recovered immediately after the end of blowdown. Thus the reflood period begins immediately after the end of blowdown.

The analysis of the reflood phase of the accident is important with regard to pipe ruptures of the reactor coolant system cold legs since the steam and entrained liquid carried out of the core for these break locations pass through the steam generators which constitute an additional energy source. The steam and entrained water leaving the core and passing through the steam generators will be evaporated and/or superheated to the temperature of the steam generator secondary fluid. The rate of energy release to the containment during the reflood phase is proportional to the core flooding rate. The rupture of the cold leg at the pump suction results in the highest mass flow through the core, and thus through the steam generators.

Mass and energy release rates during the core reflood phase of the accident, when the core is re-filling with water, were calculated by Combustion Engineering using a hydraulic resistance model and an energy balance model. The hydraulic model determines the core flooding rate. The energy balance model calculates the core exit conditions and the energy addition from the steam generator. The entrainment fraction is based on the results of the FLECHT (full length emergency core heat transfer) experiments which indicate that the fraction of fluid leaving the core during reflood is about 80 percent of the incoming flow to the core. Liquid entrainment continues until the fuel is recovered with water to about the 8-foot elevation, at which time the fuel clad temperature transient ceases (i.e., quenching occurs). Combustion Engineering has conservatively assumed quenching of the core at the 10-foot elevation for the containment functional design calculations.

Data from steam-water mixing tests conducted under joint sponsorship of the Commission and Combustion Engineering are described in Topical Reports CENPD-63, "1/5 Scale Intact Loop Post-LOCA Steam Relief Tests," dated March 1973, and CENPD-101, "Steam-Water Mixing Test Program Task D, Formal Report for Task B and Final Report for the

712 258

Steam Relief Phases of the Test Program," dated October 1973. These reports indicate that mixing will occur in the intact reactor loop between steam and the emergency core cooling system water. This mixing will act to condense some or all of the steam flowing to the containment and result in a lower containment pressure. Combustion Engineering has conservatively accounted for steam-water mixing only during the early portion of the reflooding period when the safety injection tanks are in operation. Subsequent quenching of steam by emergency core cooling system water is not assumed.

The rate of steam flow from the reactor to the containment during the reflooding period is dependent on the containment pressure. This is because the hydraulic resistance to steam flow in the reactor loops decreases with containment pressure. Combustion Engineering has selected a high containment pressure (55 pounds per square inch, gauge) for analysis of the reflood transient to maximize steam flow to the containment. The value of 55 pounds per square inch, gauge was selected to exceed the maximum containment pressure of any current plant using a reactor system designed by Combustion Engineering. The mass and energy calculations will, therefore, be conservative for plants with a calculated containment pressure less than 55 pounds per square inch, gauge. For any future plants, using the System 80 design, with a higher calculated pressure than 55 pounds per square inch, gauge, we may require additional analyses.

Combustion Engineering has calculated mass and energy release to the containment during the reflood phase of the accident using the FLOOD computer code, the same code as used by the staff. We have made comparative analyses which indicate equivalent predictions of energy release. Therefore, we have accepted Combustion Engineering's computer model as a conservative method of analysis for this plant.

Combustion Engineering has included consideration of a possible additional energy release to the containment during the post-reflood phase of the large break accident. The post-reflood phase begins after the core has been recovered with water. During this phase, decay heat generation will produce boiling in the core, and a two-phase mixture of steam and water will exist in the core. The calculations performed by Combustion Engineering assumed that this two-phase mixture rises above the core and enters the steam generator. By this process, the remainder of the available steam generator energy is removed by boiling of the water entrained in the two-phase mixture and carried into the containment as steam. In calculating the rate of energy removed from the steam generators, Combustion Engineering has used the maximum steam flow based on the hydraulic resistance of the system and steam generator heat transfer. We have reviewed Combustion Engineering's calculational method and conclude that the energy release to the containment resulting from loss-of-coolant accidents has been calculated in a conservative manner.

Combustion Engineering has calculated mass and energy release to the containment that would result from the postulated failure of a main steam line. Following rupture, steam will flow into the containment from both steam generators. Flow from the steam generator in the unbroken loop will be terminated following a main steam isolation signal and closure of the main steam isolation valve in each loop. This valve was assumed to close in 5 seconds, and the feedwater system was assumed to isolate in 20 seconds following an actuation signal. These assumptions are based on typical isolation times for pressurized water reactors.

Flashing liquid as well as heat flow from the primary system will cause the steam generator fluid level to rise following rupture. If steam is formed within the secondary fluid faster than the steam removal rate, the two-phase level will rise within the steam generators and flow through the broken steam pipe into the containment. The maximum energy release to the containment will occur if the two-phase level remains below the exit pipe so that only steam flows into the containment. For this condition the maximum amount of primary system energy will be utilized in producing steam.

Combustion Engineering has calculated the mass and energy release to the containment using the SGN-III computer code described in Appendix 6B to the CESSAR. This code calculates heat flow from the primary system into the steam generators as well as steam-water separation within each generator and entrained liquid carryover out the break. Combustion Engineering has compared the results of the SGN-III code with experimental test data. These data include blowdown of a simulated steam generator utilizing a steam separator similar to those to be used in the CESSAR System 80 design. The steam-water separation model in the SGN-III code was adjusted to yield more conservative results (higher break quality) than the test data.

Combustion Engineering has also compared the SGN-III code with the test results for other experimental facilities, (i.e., the data contained in (1) Battelle Northwest Laboratory Report BNWL 1463, "Coolant Blowdown Studies of a Reactor Simulator Vessel Containing a Perforated Sieve Plate Separator," dated February 1971; (2) General Electric Topical Report NEDO 10329, "Loss-of-Coolant Accident and Emergency Core Cooling Models for GE Boiling Water Reactors," dated April 1971; and (3) General Electric Topical Report APED-4784, "Design and Operating Experience of the ESADA Vallicitos Experimental Superheat Reactor (EVESR)," dated February 1965). The SGN-III code in each case predicted conservative results in comparison with the test data. We therefore conclude that the SGN-III code as used in the CESSAR is a conservative method for analysis of secondary system ruptures for the purposes of containment pressure analysis.

The amount of energy release to the containment from a steam line break varies with power level and break size. Combustion Engineering has analyzed spectrums of break sizes at various power levels. A break area which is 25 percent of the full pipe cross section at the hot shutdown condition was found to produce the maximum energy release to the containment.

Combustion Engineering's model did not include the additional energy release that would result from feedwater stored in the lines between the isolation valves and the steam generator inlet nozzles. The design of these lines is provided by the balance of plant designer and the length and energy content may vary. Combustion Engineering has estimated the maximum energy release to the containment associated with the fluid in these lines to be 37 million British thermal units. The resulting total energy release of 376.1 million British thermal units for the steam line break is considerably less than that from the pump suction loss-of-coolant accident which is 495.3 million British thermal units over about the same time period. Combustion Engineering indicates that the main steam line break will not be the design basis accident for the containment.

For each plant utilizing the CESSAR analysis for main steam line breaks, we will require that sufficient margin be demonstrated between the maximum containment pressure and the design pressure to provide for any additional energy release from main and auxiliary feedwater operation and the fluid stored in the feedwater lines. We will also examine the design of . . plant dependent components such as isolation valve closure times and feedwater enthalpy to determine if they are consistent with the assumptions of the CESSAR.

Combustion Engineering has calculated the mass and energy release to the containment for the short-term period following a loss-of-coolant accident for use in analysis of pressure increases in the various containment building compartments (Tables 6.2.1-31 through 6.2.1-37 in the CESSAR). Typical compartments are the reactor cavity and the compartment formed by the steam generator shield walls. The designs of these compartments will be provided by the user in its application; therefore, the adequacy of these compartments will be reviewed for each plant utilizing the CESSAR System 80 design.

The CEFLASH-4A code, which we have accepted for emergency core cooling system analysis purposes is used to calculate these mass and energy release rates. Combustion Engineering has made further conservative assumptions which act to maximize the mass and energy release rates to the containment. We conclude that the method described by Combustion Engineering will produce conservative mass and energy release rates for subcompartment analysis. For a particular subcompartment design, the use of the mass and energy data presented in the CESSAR may not be appropriate. For example, the subcompartment design and piping restraints may preclude occurrence of the full size piping breaks analyzed in the CESSAR. In the event that pipe restraints utilized for other design features of the balance of plant invalidate the break sizes and locations analyzed in the CESSAR, we will require appropriate justification as well as the associated mass and energy release to be presented in the user's application.

We will require that the methodology for calculating subcompartment mass and energy release from secondary system ruptures be presented for our review in each user's application.

6.2.2 Containment Isolation System

The containment isolation system is designed to isolate the containment atmosphere from the outside environment under accident conditions. Only those containment isolation provisions pertaining to the System 80 systems were evaluated. The detailed description of isolation provisions for the balance of plant will be supplied in each user's application. Reactor building penetration piping in the System 80 systems, up to and including the external isolation valve, will be designed as seismic Category I equipment, and will be protected against missiles that could be generated under accident conditions. Double barrier protection, in the form of closed systems and isolation valves, will be provided so that no single failure results in loss of containment integrity.

Combustion Engineering initially proposed that containment isolation be actuated on high containment pressure only. This containment isolation provision was not acceptable to us as it did not provide for containment isolation under all circumstances, (e.g., small piping break accidents). We therefore requested Combustion Engineering to justify this design and to present additional information on this subject. Amendment 40 to the CESSAR states that either low pressurizer pressure or high containment pressure will initiate containment isolation. This approach provides the required redundancy and diversity to ensure containment isolation for the postulated conditions. We therefore conclude that the containment isolation system design is acceptable.

A main steam isolation signal will occur on containment high pressure or low steam pressure. Following receipt of a containment isolation signal, all fluid penetrations not required for operation of the engineered safety features equipment will be isolated. Remotely operated isolation valves will have position indication in the control room.

We have reviewed that portion of the proposed containment isolation system design within the scope of the System 80 design for conformance to the Commission's General Design Criteria 55, 56, and 57 and for the provision for testing in accordance with Appendix J of 10 CFR Part 50. We have concluded that the proposed design, when mated with a balance of plant design that incorporates the design interface requirements provided in the CESSAR, meets the intent of the Commission's General Design Criteria and that the design will be capable of being tested in accordance with Appendix J of 10 CFR Part 50. On the basis of the review discussed above, we have concluded that the System 80 portion of the containment isolation system design is acceptable.

The fuel transfer tube closure surrounding the transfer tube utilizes a blind flange fitted with a double O-ring seal. The transfer tube closure is designed to withstand the forces resulting from a safe shutdown earthquake. Prior to returning to operation after each refueling, the leak tightness of the closure will be tested by pressurization between the two O-rings. During our review of this closure, Combustion Engineering took the position that this transfer tube is not a piping system penetration but is instead a containment access port. We agree with Combustion Engineering and consider the fuel transfer tube to be a containment access port and a part of the containment in the same sense that the equipment hatch and personnel access ports are part of the containment. We have concluded that the design meets the intent of the Commission's General Design Criterion 53 and is acceptable.

6.2.3 Containment Pressure Response for Emergency Core Cooling Evaluation

Appendix K to 10 CFR Part 50 requires that the effect of operation of all the containment installed pressure reducing systems and processes be included in the emergency core cooling system evaluation. For the purpose of emergency core cooling system evaluation it is conservative to minimize the containment pressure which increases the resistance to steam flow in the reactor coolant loops and reduces the reflood rate in the core.

Following a loss-of-coolant accident, the pressure in the containment building will be increased by the addition of steam and water from the primary reactor system to the containment atmosphere. Subsequently, following the initial blowdown, heat flow from the core, primary metal structures, and steam generators to the emergency core cooling system water, will produce additional steam. This steam together with any emergency core cooling system water spilled from the primary system will flow through the postulated break and into the containment. This energy will be released to the containment during both the blowdown and later emergency core cooling system operational phases; i.e., reflood and post-reflood.

Energy removal occurs within the containment by several means. Condensation on both the containment walls and internal structures serves as a passive energy heat sink that becomes effective early in the blowdown transient. Subsequently, the operation of the containment heat removal systems such as containment sprays and fan coolers will remove steam from the containment atmosphere. When the steam removal rate exceeds the rate of steam addition from the primary system, the containment pressure will decrease from its maximum value.

The emergency core cooling system containment pressure calculations for the CESSAR System 80 were done using the Combustion Engineering emergency core cooling system evaluation model. The Commission staff reviewed the Combustion Engineering model and published a Status Report on October 15, 1974, and amended the Status Report on November 13, 1974. We concluded that the Combustion Engineering containment pressure model was acceptable for emergency core cooling system evaluation. We required, however, that justification of the plant-dependent input parameters used in the analysis be submitted for our review of each plant. Therefore, our conclusion that the Combustion Engineering containment pressure model is acceptable for emergency core cooling system evaluation is limited to the Preliminary Design Approval stage of our review.

Containment input data were submitted in Amendment 31 to the CESSAR. Combustion Engineering included assumptions for the containment net free volume, passive heat sinks, and operation of the containment heat removal systems with regard to the conservatism for emergency core cooling system analysis. Data for the passive heat sinks were selected from a prescription which we recommended for construction permit applications. This prescription has been compiled from measurements within the containments of similar nuclear plants, as contained in Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation."

For each plant referencing the CESSAR emergency core cooling system evaluation, we will require a comparison of the significant containment parameters with those used in CESSAR. At the operating licensing stage we will require a comparison of the containment passive heat sink assumptions used in this analys.s to those that exist in the plant.

We have concluded that the plant-dependent information used for the emergency core cooling system containment pressure analysis in the CESSAR is reasonably conservative and, therefore, the calculated containment pressures are in accordance with Appendix K to 10 CFR Part 50 of the Commission's regulations.

6.3 Emergency Core Cooling System

6.3.1 Design Basis

The basic design and layout of the emergency core cooling system for CESSAR System 80 plants will be functionally similar to that developed for other Combustion Engineering plants, such as Calvert Cliffs Units 1 and 2 (Docket Nos. 50-317 and 318) and San Onofre Units 2 and 3 (Docket Nos. 50-361 and 362). The only difference is that Calvert Cliffs Units 1 and 2 and San Onofre Units 2 and 3 have three high pressure safety injection pumps (one spare) whereas CESSAR System 80 plants will have two. All three plants will have two low pressure headers. In CESSAR System 80 plants each low pressure safety injection header will feed two cold legs, whereas in Calvert Cliffs Units 1 and 2 and San Onofre Units 2 and 3, each low pressure header feeds all four cold legs.

The emergency core cooling system will be designed to provide emergency core cooling for postulated accidents where it is assumed that a failure in the reactor coolant system piping results in loss-of-coolant from the system greater than the makeup capacity of the charging pumps. The emergency core cooling system subsystems to be provided are of such number, diversity, reliability, and redundancy that no single failure of emergency core cooling system equipment occurring during a loss-of-coolant accident will result in inadequate cooling of the reactor core. Each of the emergency core cooling system subsystems will be designed to function over a range of reactor coolant system pipe break sizes, up to and including the flow area associated with a postulated double-ended break in the largest reactor coolant pipe. The emergency core cooling system will also be designed to protect against steam line break consequences.

Amendments 31, and 39-42 to the CESSAR presented analyses of the emergency core cooling system pursuant to the Final Acceptance Criteria set forth in § 50.46 and Appendix K of 10 CFR Part 50. Our review of the emergency core cooling system contained in the CESSAR evaluated (1) the loss-of-coolant accident analysis, (2) specific areas of minimum containment pressure, (3) the conformance with the single failure criterion, (4) the effects of boron precipitation on long term cooling capability, and (5) submerged valves within containment.

712 264

6.3.2 System Design

The CESSAR System 80 emergency core cooling system will consist of safety injection tanks, a high pressure injection subsystem, a low pressure injection subsystem, and a provision for recirculating flow from the containment sumps. Initially, recirculation from the containment sumps (up to two hours after the loss-of-coolant accident) will be carried out using high pressure safety injection pumps, then both high pressure and low pressure safety injection pumps will be used to meet the long term cooling requirements. Various combinations of these subsystems will assure core cooling for the complete range of postulated break sizes.

In the event of a loss-of-coolant accident, the emergency core cooling system will operate initially in the injection and subsequently in the recirculation mode. In the injection mode, high pressure safety injection will be provided by two high pressure safety injection pumps. Each pump will be sized to deliver saturated water at a rate sufficient to maintain level in the reactor vessel, matching boiloff at the time the safety injection system switches into the recirculation mode (not less than 20 minutes after the loss-of-coolant accident), assuming 25 percent spillage. The high pressure pumps will be required (balance of plant design interface requirement) to be supplied with emergency power, one pump from each of two diesel generators. Each of the injection lines will be provided with a check valve and a motor operated stop valve to isolate this subsystem from the reactor coolant system. Opening of these stop valves will be actuated by the safety injection actuation signal. The pumps will take their suction initially from the borated water in the refueling water tank and borated water will be recirculated from the containment sumps. A design requirement of the refueling water tank will be that it has sufficient capacity for at least 20 minutes of delivery at the full capacity of all safety injection and containment spray pumps after an accident. When a low level is reached in the refueling water tank, a low level signal will generate a recirculation actuation signal which will automatically transfer the pump suction to the containment sumps. Operator action will close the valves at the outlet of the refueling water tank. In the event the operator fails to close the valves from the refueling water tank, check valves will prevent backflow into the refueling water tank (see also Section 7.3.2). During the recirculation mode, the spray water will be cooled by the shutdown cooling heat exchangers prior to discharge into the containment. We have reviewed the procedure for transfer from the refueling water tank to recirculation from the containment sumps and found it acceptable.

Four safety injection tanks, each with a total volume of 2400 cubic feet and each containing a minimum of 1790 cubic feet of borated water, are provided to reflood the core during the initial stages of a loss-of-coolant accident involving large pipe breaks. Adequate water will be contained in the tanks to accomplish this function with one tank discharging through the break. Each tank will be connected to one of the cold legs of the reactor coolant system by a line with two check valves and a normally open, remotely operated isolation valve in series. The safety injection tank will, therefore, inject water automatically when the pressure in the reactor coolant system falls below the safety injection tank pressure.

During normal operation, the remotely operated valve will be maintained in the open position, and the check valves will prevent high pressure reactor coolant from flowing into the lower pressure safety injection tanks. In response to our concern that the shutdown cooling system might be exposed to safety injection tank pressures in excess of the shutdown cooling system design capability, as a result of a single failure or operator error, Combustion Engineering modified its proposed operating procedures and design of the safety injection tank isolation to (1) reduce the safety injection tank pressure from 600 to 400 pounds per square inch, gauge by operator action when reactor coolant system pressure drops to approximately 600 pounds per square inch, gauge during cooldown, and keep the isolation valve between the safety injection tank and reactor coolant system open during reactor shutdown operations until the reactor coolant system pressure has been reduced to approximately 400 pounds per square inch, gauge; (2) provide interlocks that utilize outputs from the pressurizer pressure measurement channels to prevent closure of the safety injection tank isolation valves whenever the reactor coolant system pressure is above 415 pounds per square inch, gauge; and (3) provide two fail-closed isolation valves in series in the nitrogen pressurization lines to the safety injections tanks so that a single failure will not result in an accidental increase in the pressure of the safety injection tanks (see also Section 7.6.5). When the reactor coolant system pressure drops to approximately 400 pounds per square inch, gauge, the safety injection tank isolation valves will be closed; however, a safety injection actuation signal will cause the valve to open. These isolation valves will also be interlocked with the pressurizer pressure measurement channels to open these valves automatically as reactor coolant pressure is increased to 500 pounds per square inch, gauge during startup. After the valves are opened, the valve switches will be locked open in the control room, and the valve motor circuit breakers will be racked out.

The low pressure injection system will consist of two pumps, each rated at 4200 gallons per minute design capacity and each required (balance of plant design interface requirement) to be supplied with emergency power from separate diesel generators. For the injection mode of operation, these pumps will also supply borated water from the refueling water storage tank. Sizing of the low pressure safety injection pumps will be governed by the shutdown cooling function.

When essentially all of the water in the refueling water storage tank has been injected, suction for the high pressure pumps will automatically be transferred to the containment sumps for the recirculation mode of operation, and the low pressure pumps will be automatically tripped. In the recirculation mode of operation, the emergency core cooling system will provide long-term core cooling by recirculating the spilled reactor coolant, the injected water, and the containment spray drainage, collected in the containment sumps, back to the reactor.

All of the emergency core cooling subsystems will be designed to accomplish their functions when operating on either offsite power or emergency (onsite) power. In the event of a loss-of-offsite power concurrent with a single failure in the emergency power supply system, the safety injection tanks (which require no electrical power), plus one high head and one low head injection pump would provide the minimum required emergency core cooling flow.

We have examined the information presented by Combustion Engineering concerning the available net positive suction head for the emergency core cooling system pumps. Combustion Engineering states that the high and low pressure pumps will be located in safeguards rooms in the lowest level of the auxiliary building. This location will maximize the available net positive suction head for safety injection pumps. We will review each user's application which utilizes the CESSAR System 80 design to ascertain that the specified interface condition on allowable remaining head losses are met. We conclude on this basis that the CESSAR System 80 design meets the intent of Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps" and is therefore acceptable.

6.3.3 Performance Evaluation

On January 4, 1974, the Commission published its decision in the rulemaking proceeding (Docket No. RM-50-1) concerning acceptance criteria for emergency core cooling systems for light-water-cooled nuclear power reactors. This decision included the new amendment to 10 CFR Part 50 which incorporates the ruling. The new ruling specified that boiling and pressurized light-water nuclear power reactors fueled with uranium oxide pellets within cylindrical Zircaloy cladding that are licensed after December 28, 1974 shall be provided with an emergency core cooling system which shall be designed such that its calculated cooling performance following a postulated loss-of-coolant accident conforms to the criteria set forth in § 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors" of 10 CFR Part 50. The new criteria include the following limits:

(1) The calculated maximum fuel element cladding temperature does not exceed 2200 degrees Fahrenheit.

(2) The calculated total oxidation of the cladding does not exceed 0.17 times 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 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding 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 emergency core cooling system, 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 remaining in the core.

In addition, § 50.46 states that emergency core cooling system cooling performance shall be calculated in accordance with an acceptable evaluation model, and shall be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered. Appendix K, "ECCS Evaluation Models," of 10 CFR Part 50, sets forth certain required and acceptable features of evaluation models.

The emergency core cooling system analysis in the CESSAR was performed with an evaluation model which conforms with Appendix K of 10 CFR 50. Our review of this model is documented in the following reports: (1) Status Report by the Directorate of Licensing in the Matter of Combustion Engineering, Inc., ECCS Evaluation Model Conformance to 10 CFR Part 50, Appendix K, October 1974; (2) Supplement to the Status Report by the Directorate of Licensing in the Matter of Combustion Engineering, Inc., ECCS Evaluation Model Conformance to 10 CFR Part 50, Appendix K, November 13, 1974; and (3) NRC Staff Review of the Combustion Engineering ECCS Evaluation Model, Letter to Combustion Engineering dated June 13, 1975.

The emergency core cooling system analysis submitted in Amendment 31 to the CESSAR was performed using the approved Combustion Engineering emergency core cooling analysis evaluation model, except for changes in three areas. Combustion Engineering submitted Topical Report CENPD-132, Supplement 2P, "Calculational Methods for the C-E Large Break LOCA Evaluation Model" dated July 1975, in support of the three proposed model changes. Our letter of December 9, 1975 to Combustion Engineering presented the results of our review and conclusions regarding the proposed changes. The staff concluded that the proposed changes in the containment wall noding and injection section pressure drop were acceptable, but that the proposed modification to the reflood heat transfer coefficient was not sufficiently justified. Therefore, a revised emergency core cooling system analysis using the approved value for reflood heat transfer was performed, and the results of this analysis were submitted in Amendment 41 to the CESSAR.

Amendment 41 to the CESSAR addresses a spectrum of nine breaks for the loss-of-coolant from major reactor coolant system pipe ruptures. Included were two analyses for hot leg and pump suction leg large breaks confirming that these breaks are not limiting. In addition, Amendment 31 to the CESSAR included analyses for a spectrum of 4 small breaks confirming that they are not limiting. The small break analyses were performed using the approved Combustion Engineering emergency core cooling system evaluation model. The worst break was identified as the double ended guillotine break located in the pump discharge and having a discharge coefficient of 1.0 The following table summarizes the emergency core cooling system calculation results for the limiting fuel rod at a linear heat generation rate of 12.1 kilowatts per foot, and for the limiting break.

Parameter	Value	Criterion Limit
Peak Clad Temperature, Degrees Farenheit	2146	2200
Maximum Local Clad Oxidation, Percent	16.05	17.0
Maximum Core-Wide Clad Oxidation, Percent	<0.923	1.0

As indicated by the table, the predicted values for peak clad temperature and local and core-wide oxidation are below the corresponding limits of 2200 degrees Farenheit, 17 percent and 1.0 percent as specified in § 50.46 of 10 CFR Part 50.

For calculational flexibility, the initial values used for fuel and clad temperature in the calculation of maximum local and core-wide clad oxidation were based on a peak linear heat generation rate of 16.0 kilowatts per foot, rather than the expected value of 12.1 kilowatts per foot. The resulting values predicted for local and core-wide clad oxidation were therefore conservatively high.

The effect of rod bow on fuel rod behavior has not been included in the emergency core cooling system analysis for the CESSAR System 80 design in an explicit manner. However, prior to issuing an Operating License to any plant referencing the CESSAR, information on rod bow for Combustion Engineering 16 x 16 fuel will be available. This information will be used to assess the effect of rod bow on emergency core cooling system performance. The operating technical specification limits established during the review of the Final Safety Analysis Report of such plants will include a consideration of rod bow effects.

Based on our review, we conclude that the emergency core cooling system performance included in the CESSAR conforms to the peak clad temperature and maximum oxidation and hydrogen generation criteria of § 50.46(b) of 10 CFR Part 50.

6.3.3.1 Minimum Containment Pressure

The plant-dependent input parameters used in the containment pressure calculations were submitted in Amendment 31 to the CESSAR. Included was a tabulation of containment mass and energy release values. The parameters used for the containment pressure calculation were conservatively determined in accordance with our prescription recommended for construction permit applications contained in the Branch Technical Position CSB 6-1. This prescription was compiled from measurements within the containments of similar nuclear plants, wherein the containment heat removal system was assumed to operate at maximum capacity and the spray water and service water temperatures were assumed to be at their minimum operational values (see also Section 6.2.3).

For each balance of plant design utilizing the CESSAR, we will require during the construction permit or balance of plant review, a comparison of the significant containment parameters with those used in the CESSAR emergency core cooling system evaluation. At the operating licensing stage we will require a comparison of the containment passive heat sink assumptions used in the CESSAR analysis to those that exist in the plant.

We have concluded that the plant-dependent information used for the emergency core cooling system containment pressure analysis in the CESSAR is conservative, and that the calculated containment pressures are in conformance with Appendix K to 10 CFR Part 50.

6.3.3.2 Single Failure Criterion

Combustion Engineering's Topical Report CENPD-123P, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," August 1974, describes an analysis of the possible single failures that can occur within the emergency core cooling system. It was concluded that the worst single failure for the large break in Combustion Engineering plants was the loss of one of the low pressure pumps. This assumption was used in the emergency core cooling system evaluation in the CESSAR. Our status report of October 1974 states that we found Combustion Engineering's generic evaluation of the single failure criterion acceptable, but that the satisfaction of the single failure criterion specified in Appendix K to 10 CFR Part 50 should be confirmed individually for each plant. The position concerning single failures for the System 80 design described in the CESSAR, and our review of this subject is summarized below.

Each motor operated and air operated valve in the emergency core cooling system has been reviewed to determine if a single malfunction of the operator will have an adverse effect on the emergency core cooling system. In each case, the valve was assumed to fail or malfunction to the most adverse position rather than the normal failed position. We have concluded that redundancy of systems and/or valves provides for proper functioning of the emergency core cooling system, with the qualifications discussed below.

(1) Safety Injection Tank Isolation Valves

To preclude loss of the safety function provided by the safety injection tanks, electric power will be removed from the safety injection tank motor-operated isolation valves while the valves are in the open position. After each valve is opened, it will be locked open in the control room, and the motor's circuit breaker will be racked out.

We will require that each plant referencing the CESSAR System 80 design include a requirement to lock out power to each safety injection tank isolation valve in the technical specifications of the Final Safety Analysis Report.

712 270

(2) Mini-flow Bypass Valves

In the event of a small break loss-of-coolant accident, the reactor coolant system pressure could remain relatively high thereby preventing flow in the low pressure safety injection pumps. If this high pressure is sustained for sufficient time, it would result in overheating and damage to the pumps. To prevent overheating and pump damage, orificed mini-flow bypass lines have been provided which allow a small flow of coolant from the discharge of each low pressure safety injection pump back to the refueling water storage tank. These mini-flow bypass lines must be open during the injection phase of a loss-of-coolant accident until the reactor coolant system pressure falls below the shutoff head of the low pressure safety injection pumps. However, these lines must be closed to allow isolation of the refueling water storage tank and containment during recirculation. For this purpose, motor operated valves have been provided in each mini-flow bypass line. The CESSAR includes interface requirements that the design of the mini-flow isolation valve system be such that any single failure would not prevent proper isolation of the refueling water storage tank during the recirculation mode, or result in loss of emergency core cooling function during the injection mode. These interface requirements will form the basis for our review of each utility-user's application. We will require that these design objectives be confirmed by each plant referencing CESSAR.

(3) Hot Leg Injection Valves

To prevent boron precipitation during long term cooling following a loss-of-coolant accident, the CESSAR proposes to supply core flushing by injecting part of the emergency core cooling injection water through the shutdown cooling lines into the hot legs. We will require confirmation that each plant referencing the CESSAR implements a hot leg injection system design such that the single failure criterion is satisfied. The CESSAR presently includes this requirement as an interface item in conformance with our requirements.

On the basis of our review, we conclude that the System 80 design as described in CESSAR is in conformance with the single failure criterion of 10 CFR 50.46. We will require that each plant referencing CESSAR satisfactorily confirm its design regarding the safety injection tank isolation valves, the mini-flow bypass valves and the hot leg injection valves.

6.3.3.3 Boric Acid Concentration Effects During Long Term Cooling

The emergency core cooling system is required to provide adequate cooling for the reactor core following a loss-of-coolant accident. Long term residual heat removal is provided by continuous evaporation of core coolant in the reactor vessel which may result in high concentration of boric acid and other materials in the vessel. If the solubility limit is exceeded, precipitation of boric acid will occur resulting in possible blockage of the coolant flow paths and a degradation in cooling capability.

To prevent boron precipitation following a loss of coolant accident, the CESSAR System 80 design includes a method of core flushing which utilizes simultaneous hot and cold leg injection from the high pressure safety injection pumps. This is accomplished by opening the hot leg injection lines from each high pressure safety injection header to the shutdown cooling suction lines. Flows will be balanced so that 50 percent of the high pressure pump flow is delivered to the hot legs and 50 percent to the cold legs. Assuming a single failure in one of the high pressure pumps, 50 percent of the flow from the remaining pump would provide about 75 pounds per second of flow to the hot legs. We have performed an independent analysis for a time of 3 hours after a loss-of-coolant accident which indicates that a hot leg injection flow of about 46 pounds per second would be required to match boil off and provide sufficient flushing to prevent boron precipitation. Our analysis also indicates that boron precipitation will not occur prior to 4 hours following a loss-of-coolant accident. Combustion Engineering has stated that the low pressure safety injection pumps would be available to provide additional hot leg injection flow, if required.

The relatively high steam velocities in the hot legs could cause entrainment of emergency core cooling water and impair hot leg injection if initiated too early.

The CESSAR proposes to initiate hot leg injection within 90 minutes after a loss-of-coolant accident. Combustion Engineering has indicated that steam velocities in the hot legs will not interfere with emergency core cooling injection at this time. Based upon our preliminary independent calculations, we concur with Combustion Engineering; however, we will require that the emergency procedures specifying the initiation of hot leg injection be finalized and be submitted with the application for Final Design Approval.

On the basis of our review, we conclude that the system design is acceptable for preventing excessive boric acid buildup in the reactor vessel, and that the long term cooling criterion of 10 CFR 50.46(b) will be satisfied.

6.3.3.4 Submerged Valves

The CESSAR delineates an interface requirement that states that flooding shall not preclude minimum acceptable recirculation capability. We find this interface requirement acceptable for Preliminary Design Approval. We will review each user's application that references or incorporates the CESSAR to ascertain that the above design objective has been met.

6.3.3.5 Evaluation Conclusions

Based on our review, we conclude that the emergency core cooling system performance for the System 80 design described in the CESSAR will conform to the peak clad temperature and maximum oxidation and hydrogen generation criteria of § 50.46(b) of 10 CFR Part 50, provided that the maximum linear heat generation rate does not exceed 12.1

712 272

kilowatts per foot. In addition, each user's application referencing CESSAR must also conform to the two remaining criteria, i.e., the maintenance of a coolable geometry and long term cooling.

We have reviewed the emergency core cooling system containment pressure analysis in the CESSAR, and we conclude that it is in conformance with § 50.46 of 10 CFR Part 50. As noted in Section 6.3.3.1 above, we will require that each user's application referencing the CESSAR confirm its plant dependent parameter assumptions.

Based on our review of the single failure criterion as it applies to the emergency core cooling system of the CESSAR System 80 design, we conclude that the criterion of Appendix K of 10 CFR Part 50 will be satisfied provided the requirements noted in Section 6.3.3.2 above are met regarding confirmation of the operating procedures and system design for the safety injection tank isolation valves, the mini-flow isolation valves and the hot leg injection valves.

We have reviewed the results of analyses and the proposed emergency core cooling system design with respect to long term cooling and the effects of boric acid concentration. We conclude that the proposed system design is acceptable and that the long term cooling criterion of § 50.46(b) of 10 CFR Part 50 will be satisfied. As noted in Section 6.3.3.3 above, we will require that more specific information regarding the operating procedures designed to prevent boron precipitation be submitted with the application for Final Design Approval.

We have reviewed the interface requirement delineated in the CESSAR regarding the possible submergence of emergency core cooling system valves within containment, and conclude that it is acceptable. We will review each user's application utilizing the CESSAR to ascertain that this interface requirement is met.

6.3.4 Tests and Inspections

Section 6.3.4 of the CESSAR states that operability of the emergency core cooling system will be demonstrated by preoperational tests of the system and by component tests in conformance with Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors." Preoperational tests will consist of:

(1) Emergency core cooling system pump net positive suction head tests for the injection mode.

(2) Net positive suction head tests for the recirculation mode in conjunction with low pressure safety injection ambient condition recirculating tests.

(3) Ambient condition flow tests for the high pressure safety injection and low pressure safety injection systems for the injection mode.

712 273

(4) Blowdown tests at reduced pressure to assure that the safety injection tanks are capable of flooding the core at the required rate.

(5) Low pressure safety injection ambient condition recirculation tests to demonstrate the ability of the low pressure safety injection pumps to operate taking suction along the recirculating flow path.

(6) Tests to show the ability of the emergency core cooling system to transfer from the injection mode to the recirculation mode.

(7) Tests to show the operability of the check valves along the safety injection discharge path at operating temperatures.

Combustion Engineering initially stated that preoperational net positive suction head tests for the recirculation mode were impractical due to a lack of a source of water to supply the containment sumps. Combustion Engineering has modified the CESSAR to state that tests would be performed in conformance with the guidance of Regulatory Guide 1.79, and that the tests would demonstrate the ability of the emergency core cooling system to transfer from the injection mode to the recirculation mode.

Component tests will be performed to verify power operation of the safety injection components. Inservice testing for all Class 2 and 3 pumps, and Class 1, 2 and 3 valves will be in accordance with the ASME Code, Section XI, Summer 1973 Addenda, Subsections IWP, and IWV, respectively. The tests include cycling of all check valves to ensure proper operation and checking of instrumentation channels vital to the emergency core cooling system operation. Conformance with the above code requirements constitutes an acceptable basis for satisfying the applicable portions of the Commission's General Design Criteria 37, 40, 43 and 46.

We have reviewed the emergency core cooling system test program described in the CESSAR and have concluded that it will adequately demonstrate the operability of the emergency core cooling system and is acceptable. We have also reviewed the proposed design and have concluded that adequate consideration has been given to design features that permit the system and components to be tested.

6.3.5 Design Interface Requirement for the Balance of Plant

The CESSAR specifies design interface requirements for the balance of plant design to assure that the assumptions concerning the balance of plant design that were made by Combustion Engineering in its design and evaluation of the CESSAR emergency core cooling system are valid, and that the emergency core cooling system will meet its functional design requirements.

712 274

We have reviewed these interface requirements and conclude that they adequately specify the balance of plant design requirements related to the emergency core cooling system and that they are acceptable.

These interface requirements will form the basis for our review of each utility-user's application that utilizes the CESSAR. Users that reference the CESSAR will be required to meet all of these specified interface requirements.

6.4 Engineered Safety Feature Materials

The mechanical properties of materials selected for the engineered safety features will satisfy Appendix I of Section III of the ASME Code, or Parts A, B, and C of Section II of the code, and the staff position that the yield strength of cold stainless steels shall be less than 90,000 pounds per square inch. We will require that interface requirements be included that assure that the controls on the hydrogen ion concentration of the reactor containment sprays following a postulated loss-of-coolant accident satisfy the staff position that the hydrogen ion concentration of the spray be adequate to ensure freedom from stress corrosion cracking of the austenitic stainless steel components and welds of the emergency core cooling systems throughout the duration of the postulated accident to completion of cleanup. The controls on the use and fabrication of the austenitic stainless steel in the CESSAR satisfy the recommendations of Regulatory Guides 1.31 and 1.44. Fabrication and heat treatment practices that will be performed in accordance with these requirements provide added assurance that stress corrosion cracking will not occur during the postulated accident time interval.

Conformance with the above codes, Regulatory Guide recommendations and staff positions on the allowable maximum yield strength of cold worked austenitic stainless steel, and the allowable range of the hydrogen ion concentration of the containment sprays constitutes an acceptable basis for meeting the requirements of the Commission's General Design Criteria 35, 38, and 41.

712 275

7.0 INSTRUMENTATION AND CONTROL

7.1 General

The Commission's General Design Criteria, the Institute of Electrical and Electronics Engineer (IEEE) Standards including IEEE Std 279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations" applicable Regulatory Guides for power reactors, and staff technical positions noted in Table 7-1 of the Commission's Standard Review Plan have been utilized as the bases for evaluating the adequacy of the protection and control systems. This safety evaluation report reflects the results of our review of the CESSAR through Amendment 44. Specific documents employed in the review are listed in Appendix B of this report.

We have reviewed the interface information provided in the CESSAR for the instrumentation and controls associated with the proposed design. We have found that the interface information and criteria contained in the CESSAR, as supplemented with interface requirements included in this report, provide reasonable assurance that the balance of plant design can be accomplished in a manner that will validate the assumptions in Section 15 of the CESSAR. Based on the above, we conclude that the instrumentation and control systems specified in the CESSAR can be implemented in an acceptable manner.

The sections that follow provide additional interface information to that of the CESSAR that we have identified, and the interface acceptance criteria are listed in Table 7-1 of this report for specific CESSAR systems. We will review the implementation of each interface requirement specified in the CESSAR and in Table 7-1 of this report during our review of each user's application that references the CESSAR to ascertain that these requirements are satisfied.

7.2 Reactor Trip System

The reactor trip system of the CESSAR will consist of four redundant and independent protection channels for each reactor trip output. Each channel will provide three inputs into three of six independent logic matrices representing all possible two-out-of-four trip combinations for the four protection channels. The six logic matrices provide four redundant and independent trip paths, such that each logic matrix can interrupt each of the four trip paths. Thus, a trip output from any one of the six logic matrices interrupts power to the control rod power supply breakers causing insertion of all rods.

The following trip inputs are part of the reactor trip system:

712 276

TABLE 7-1 - INTERFACE ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS

CRITERIA	TITLE	SAR/SER SECTION 7.2 RTS	7.3 ESFS				
		ALL TRIPS	ECCS	MAIN STEAM ISOLATION SYS. (BOP)	ESFAS	CONTAINMENT SYSTEMS (BOP)	SERVICE/ COOLING WATER SYSTEMS (BOP)
10 CFR PART 50	CONTENTS OF APPLICATION TECHNICAL INFORMATION	X	X	X	X	X	X
	TECHNICAL SPECIFICATIONS	X	X	X	X	X	X
	CODES AND STANDARDS	X	X	X	X	X	X
GENERAL DESIGN CRITERIA (GDC) APPENDIX A TO 10 CFR PART 50	(SEE STANDARD REVIEW PLAN TABLE 7-1 FOR SPECIFIC GDC & TITLE)	X	X	X	X	X	X
IEEE STANDARDS IEEE STD 279-1971 (ANSI N42.7-1972)		X	X	X	X	X	X
IEEE STD 308-1971			X	X	X	X	X
IEEE STD 317-1972		X	X	X	X	X	
IEEE STD 323-1974		X	X	X	y	X	X
IEEE STD 336-1971		X	X	X	X	X	X
IEEE STD 338-1971		X	X	X	X	X	X
IEEE STD 344-1975		X	X	X	X	X	X
IEEE STD 379-1972		X	X	X	X	X	X
IEEE STD 384-1974		X	X	X	X	X	X
REGULATORY GUIDES (RGS) RG 1.6		X	X	X	X	X	X
RG 1.11		X	X	X	X	X	
RG 1.22		X	X	X	X	X	X
RG 1.29		X	X	X	X	X	X
RG 1.30		X	X	X	X	X	X
RG 1.32			X	X	X	X	X
RG 1.47		X	X	X	X	X	X
RG 1.53		X	X	X	X	X	X
RG 1.62		X	X	X	X	X	X
RG 1.63		X	X	X	X	X	
RG 1.68		X				X	
RG 1.70		X	X	X	X	X	X
RG 1.75		X	X	X	X	X	X
RG 1.89		X	X	X	X	X	X
BRANCH TECHNICAL POSITIONS (BTP) BTP EICSB 1		X	X	X	X	X	X
BTP EICSB 3							
BTP EICSB 4			X				
BTP EICSB 5		X	X				
BTP EICSB 9		X	X	X	X	X	X
BTP EICSB 10		X	X	X	X	X	X
BTP EICSB 12		X					
BTP EICSB 13					X		
BTP EICSB 14		X					
BTP EICSB 15		X					
BTP EICSB 18			X	X		X	X
BTP EICSB 20			X		X		
BTP EICSB 21		X	X	X	X	X	X
BTP EICSB 22		X	X	X	X	X	X
BTP EICSB 23							
BTP EICSB 24		X	X	X	X	X	X
BTP EICSB 25			X				
BTP EICSB 26		X					
BTP EICSB 27			X	X	X	X	X

TABLE 7-1 - INTERFACE ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS

CRITERIA	TITLE	SAR/SER SECTION 7.4 SYSTEMS FOR SAFE SHUTDOWN			
		EMERGENCY FEEDWATER SYSTEM (BOP)	ATMOSPHERIC DUMP SYSTEM (BOP)	SHUTDOWN COOLING SYSTEM	BORON ADDITION PORTION OF CVCS
10 CFR PART 50	CONTENTS OF APPLICATION TECHNICAL INFORMATION	X	X	X	X
	TECHNICAL SPECIFICATIONS	X	X	X	X
	CODES AND STANDARDS	X	X	X	X
GENERAL DESIGN CRITERIA (GDC) APPENDIX A TO 10 CFR PART 50	(SEE STANDARD REVIEW PLAN TABLE 7-1 FOR SPECIFIC GDC & TITLE)	X	X	X	X
IEEE STANDARDS					
IEEE STD 279-1971 (ANSI N42.7-1972)		X	X	X	X
IEEE STD 308-1971		X	X	X	X
IEEE STD 317-1972				X	X
IEEE STD 323-1974		X	X	X	X
IEEE STD 336-1971		X	X	X	X
IEEE STD 338-1971		X	X	X	X
IEEE STD 344-1975		X	X	X	X
IEEE STD 379-1972		X	X	X	X
IEEE STD 384-1974		X	X	X	X
REGULATORY GUIDES (RGS)					
RG 1.6		X	X	X	X
RG 1.11				X	X
RG 1.22		X	X	X	X
RG 1.29		X	X	X	X
RG 1.30		X	X	X	X
RG 1.32		X	X	X	X
RG 1.47		X	X	X	X
RG 1.53		X	X	X	X
RG 1.62		X			
RG 1.63				X	X
RG 1.68					X
RG 1.70		X	X	X	X
RG 1.75		X	X	X	X
RG 1.89		X	X	X	X
BRANCH TECHNICAL POSITIONS (BTP)					
BTP EICSB 1		X	X	X	
BTP EICSB 3				X	
BTP EICSB 4					
BTP EICSB 5					
BTP EICSB 9		X	X	X	X
BTP EICSB 10		X	X	X	X
BTP EICSB 12					
BTP EICSB 13		X			
BTP EICSB 14					
BTP EICSB 15					
BTP EICSB 18		X	X	X	X
BTP EICSB 20					
BTP EICSB 21		X	X	X	X
BTP EICSB 22		X	X	X	X
BTP EICSB 23					
BTP EICSB 24		X	X	X	X
BTP EICSB 25					
BTP EICSB 26					
BTP EICSB 27		X	X	X	X

TABLE 7-1 - INTERFACE ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS

CRITERIA	TITLE	7.5 SAFETY RELATED DISPLAY INSTRUMENTATION	7.6 OTHER SYSTEMS REQUIRED FOR SAFETY				7.7 CONTROL SYSTEMS
		POST-ACCIDENT MONITORING SYSTEM	SCS (RHS) ISOLATION VALVES	SAFETY INJECTION TANK ISOLATION VALVES	INSTRUMENTATION AND CONTROL POWER SUPPLY		
10 CFR PART 50	CONTENTS OF APPLICATION TECHNICAL INFORMATION	X	X	X	X		X
	TECHNICAL SPECIFICATIONS	X	X	X	X		X
	CODES AND STANDARDS	X	X	X	X		X
GENERAL DESIGN CRITERIA (GDC) APPENDIX A TO 10 CFR PART 50	(SEE STANDARD REVIEW PLAN TABLE 7-1 FOR SPECIFIC GDC & TITLE)	X	X	X	X		X
IEEE STANDARDS							
IEEE STD 279-1971 (ANSI N42.7-1972)		X	X	X	X		X
IEEE STD 308-1971		X	X	X	X		
IEEE STD 317-1972		X	X	X	X		X
IEEE STD 323-1974		X	X	X	X		
IEEE STD 336-1971		X	X	X	X		X
IEEE STD 338-1971		X	X	X	X		
IEEE STD 344-1975		X	X	X	X		
IEEE STD 379-1972		X	X	X	X		X
IEEE STD 384-1974		X	X	X	X		X
REGULATORY GUIDES (RGS)							
RG 1.6		X	X	X	X		
RG 1.11		X	X	X			
RG 1.22		X	X	X	X		
RG 1.29		X	X	X	X		
RG 1.30		X	X	X	X		X
RG 1.32		X	X	X	X		
RG 1.47		X	X	X	X		
RG 1.53		X	X	X	X		X
RG 1.62							
RG 1.63		X	X	X	X		X
RG 1.68		X					
RG 1.70		X	X	X	X		X
RG 1.75		X	X	X	X		X
RG 1.89		X	X	X	X		
BRANCH TECHNICAL POSITIONS (BTP)							
BTP EICSB 1					X		
BTP EICSB 3			X				
BTP EICSB 4				X			
BTP EICSB 5							
BTP EICSB 9		X	X	X			
BTP EICSB 10		X	X	X			
BTP EICSB 12							
BTP EICSB 13							
BTP EICSB 14							X
BTP EICSB 15							
BTP EICSB 18				X			
BTP EICSB 20							
BTP EICSB 21		X	X	X	X		
BTP EICSB 22		X	X	X	X		
BTP EICSB 23		X					
BTP EICSB 24			X	X			
BTP EICSB 25							
BTP EICSB 26							
BTP EICSB 27			X	X			

(1) High linear power level

(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) Steam generator 1 water level low

(8) Steam generator 1 pressure low

(9) Steam generator 1 water level high

(10) Steam generator 2 water level low

(11) Steam generator 2 pressure low

(12) Steam generator 2 water level high

(13) High containment pressure

The following sections address the problem areas revealed during our review, and their resolutions.

7.2.1 Computer Protection System

Combustion Engineering initially proposed the use of computer-based systems for implementing the high local power density and low departure from nucleate boiling ratio trip functions. The proposed computer protection system will consist of four redundant digital computers, identified as core protection calculators. They will acquire data from plant process sensors and from two redundant computer-based control element assembly calculators which will provide each core protection calculator with control rod position deviation information. Each core protection calculator will provide trip inputs to one of the four redundant and independent reactor trip system channels when the trip setpoints for high local power density and/or low departure from nucleate boiling ratio are exceeded. The hardware configuration block diagram in Figure 7-1 of this report depicts functionally the scope of the core protection calculator including its interactions with the reactor trip system, the computer-based core operating limit supervisory system and the plant computer.

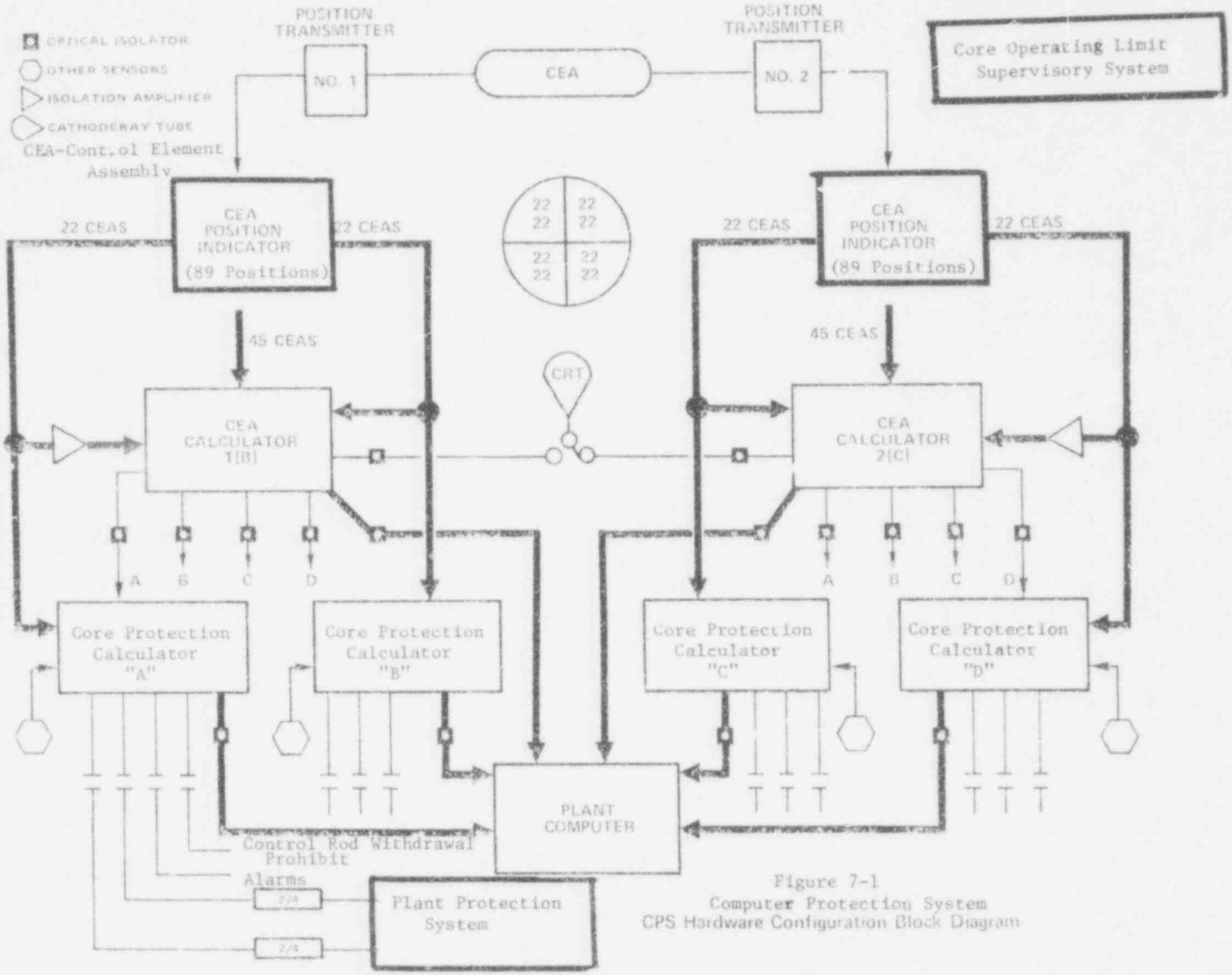


Figure 7-1
Computer Protection System
CPS Hardware Configuration Block Diagram

712 281

The core operating limit supervisory system will be utilized to assure that the operator maintains the reactor within the conditions assumed in the safety analysis (also see Sections 4.3, 7.7 and 15.3 for discussion of core operating limit supervisory system).

Our review of the computer protection system identified many areas both for the hardware and software which require in depth review. We also noted that the core protection system presented in the CESSAR is identical to that of the Arkansas Nuclear One, Unit 2 (Docket No. 50-368) currently undergoing review for an operating license. Following discussions with the staff, Combustion Engineering documented in the CESSAR that the computer protection system of System 80 will be considered as an item of research and development for which the design will be pursued on the Arkansas Nuclear One, Unit 2 application. Also, Combustion Engineering indicated that in the event the development program results prove to be unacceptable to the staff, an alternate design such as that provided for this function on the St. Lucie Unit 1 facility (Docket No. 50-335) which has been reviewed and approved, will be substituted.

With regard to the St. Lucie Unit 1 backup design, the Commission staff has reviewed and found acceptable the reactor trip system for this facility. Since the analog portion of the reactor trip system proposed in the CESSAR is compatible with that of St. Lucie Unit 1, it is our judgment that the St. Lucie Unit 1 reactor trip system design can satisfactorily be implemented as the backup design for CESSAR System 80 Design insofar as it relates to the electrical, instrumentation and control aspects of the design.

We conclude that the commitment to make the computer protection system an item of research and development and to provide a backup design is acceptable and satisfies our present evaluation requirements. The review of the computer protection system is being accomplished on a generic basis, using the Arkansas Nuclear One, Unit 2 final core protection system design as the base. The results of our review of the core protection system will be reported prior to or in the Safety Evaluation Report for Arkansas Nuclear One, Unit 2 which is scheduled to be issued in mid-1976. Thus, a resolution to this issue will be available well before the CESSAR System 80 Final Design Approval application is received.

7.2.2 Equipment Protection Trips

Combustion Engineering identified the loss-of-load trip and its bypass, and high steam generator water level reactor trip functions as being required for equipment protection and not for plant safety. Combustion Engineering indicated that because no credit is taken for these functions in the safety analysis, these trips do not have to satisfy IEEE Std 279-1971. We advised Combustion Engineering that the introduction of any trip into the reactor trip system should be accomplished in a manner that will not degrade the reactor system, and that we require that the loss-of-load trip and bypass and the high steam generator water level trip be designed to satisfy IEEE Std 279-1971. Combustion Engineering has documented in the CESSAR that the high steam generator level trip will conform with our position and it is, therefore, acceptable.

With regard to the loss of load trip and bypass, Combustion Engineering had agreed to design the trip to satisfy IEEE Std 279-1971 except for the seismic qualification of the channel sensors. The loss of load trip bypass would not meet IEEE Std 279-1971. As the sensors will be located in the non-seismic Category I turbine area and their failure could not degrade the reactor trip system, we consider the exception to seismically qualifying the loss-of-load trip sensors acceptable. Requests for exemption from the requirement to conform to IEEE Standard 279 will be considered in applications referencing the CESSAR. Concerning the loss-of-load trip bypass, we advised Combustion Engineering that the proposed single bypass to inhibit the four trip channels would compromise the independence of the reactor trip system and that the design was unacceptable. We informed Combustion Engineering that we would require that either (1) the loss-of-load trip be deleted, or (2) the bypass feature be removed, or (3) an alternate design be submitted for our review which would maintain the inherent independence of the reactor trip system channels. Combustion Engineering has subsequently elected to delete the loss-of-load trip and bypass. We consider this acceptable.

7.2.3 Interface Requirements

Our review of the reactor-trip system revealed the following additional interface requirements that will need to be satisfied in the balance of plant design:

(1) Four physically and electrically independent 125 volt direct current sources shall be supplied for the reactor trip breakers.

(2) Each source of auxiliary alternating current power and standby onsite power shall be physically and electrically independent as required by General Design Criterion 17.

(3) The balance-of-plant design shall satisfy all the interface requirements listed for the reactor trip system in Table 7-1 of this report.

7.2.4 Conclusions

We reviewed the reactor trip system description, including functional logic diagrams, testing capabilities, control of bypasses, interface requirements, Combustion Engineering's proposed design criteria and design bases, and Combustion Engineering's analysis of the adequacy of those criteria and bases.

In applications referencing the CESSAR and having a balance of plant design for the reactor trip system that satisfies the interface requirements stated in Section 7.2.3 above, there is reasonable assurance that the reactor trip design proposed in the CESSAR will satisfy the Commission's requirements stated in Section 7.1 of this report.

On the basis of our review, and the interface requirements specified in the CESSAR, and as supplemented by this report, we have concluded that the reactor trip system can meet the Commission's requirements and is acceptable.

7.3 Engineered Safety Features Actuation Systems

The engineered safety feature systems are initiated and controlled by the engineered safety feature actuation systems. Each actuation system is identical except for the input parameters and includes four redundant and independent protection channels per trip input. Each actuation system logic is configured in the same manner as for the reactor trip system with the four trip path outputs arranged into two independent, selective, two-out-of-four coincidence logics, each logic serving one redundant group of engineered safety feature equipment.

Engineered safety feature system actuation signals and associated trip inputs identified in CESSAR are as follows:

(1) Containment isolation actuation signal; high containment pressure, or low pressurizer pressure

(2) Safety injection actuation signal; low pressurizer pressure, or high containment pressure

(3) Containment spray actuation signal; high-high containment pressure and (a) low pressurizer pressure, or (b) high containment pressure

(4) Main steam isolation signal; low pressure in either of the two steam generators, or high containment pressure

(5) Recirculation actuation signal; low refueling tank water level

(6) Emergency feedwater actuation signal; the trip inputs associated with this actuation signal will be supplied in the balance of plant design.

The only engineered safety feature system actuated by the actuation systems within the System 80 scope of design is the emergency core cooling system. Thus, design compatibility could only be established between the actuated emergency core cooling system and actuating safety injection actuation signal and recirculation actuation system. The emergency feedwater system actuation signal as well as the actuated systems are outside the System 80 design scope and they will be evaluated during the review of user's applications referencing System 80. The remaining containment isolation actuation signal, containment spray actuation signal and main steam isolation signal have been evaluated on the basis that the actuated engineered safety feature system is composed of two redundant and independent trains of components and systems configured as required so the system safety function can be accomplished, assuming a single failure.

The following sections address the problem areas revealed during our review and the resolutions concerning them.

7.3.1 Steam Line Break Isolation

In the analysis of the steam line break accident inside the containment, Combustion Engineering takes credit for the termination of main feedwater flow to the affected steam generator and the timely isolation of the intact steam generator. Those two actions are accomplished by the main steam isolation system which effects the closure of the main steam and feedwater isolation valves.

Our review of the typical preliminary design revealed that assumptions made in this accident analysis would be invalidated by a single failure in the main feedwater isolation valves. Combustion Engineering subsequently documented in CESSAR an interface requirement that specifies two isolation valves in series in each of the main feedwater lines to assure the termination of main feedwater flow when required, assuming a single failure. With this interface and based on our review of the System 80 design in this regard, there is reasonable assurance that the instrumentation, control and electrical equipment associated with the main feedwater line isolation valves can be designed to satisfy the Commission's requirements and, therefore, is acceptable.

With regard to the isolation of the intact steam generator at the steam side, our review revealed two areas of concern in the event of a steam line break inside the containment:

(1) Combustion Engineering has indicated that credit is taken for the closure of all steam paths downstream of the main steam isolation valve when a single failure prevents the closure of the main steam isolation valve or the main steam isolation valve bypass in the intact line from the steam generator. Since the steam line break is considered one of the major design basis accidents, either the consequences of this accident must be demonstrated to be acceptable, or the structures, systems, and components provided to make the consequences acceptable shall be designed to conform with the requirements set forth in Appendix A of 10 CFR Part 50 and Appendix A of 10 CFR Part 100. The balance of plant design shall either demonstrate that the consequences of two steam generators blowing down are acceptable, or provide a design which meets the single failure criterion. Based on this interface requirement and our review of the CESSAR, there is reasonable assurance that the instrumentation, control and electrical equipment associated with the isolation of all steam paths downstream of the main steam isolation valves can be made to satisfy the Commission's requirements, and therefore is acceptable.

(2) The System 80 design requires that each steam generator be provided with two air-operated atmospheric dump valves arranged in parallel and located between the steam generator and the main steam isolation valve. It is our concern that in the event of a steam line break accident inside the containment, a single failure causing either of these valves to open or preventing one of them in the

line from the intact steam generator from closing could result in the blowing down of both steam generators. It is our position that the balance of plant design shall satisfy the interface requirement to demonstrate that the consequences of a steam line break with a failure of either atmospheric dump valve are acceptable, or provide a design which meets the single failure criterion and satisfies the Commission's requirements. In addition, the balance of plant design shall satisfy all the interface criteria listed for the main steam isolation system in Table 7-1 of this report. With these interface requirements and based on our review of the System 80 design in this regard, there is reasonable assurance that the instrumentation, control and electrical equipment associated with the atmospheric dump valve can be made to satisfy the Commission's requirements and, therefore, it is acceptable.

7.3.2 Refueling Water Tank Isolation During Recirculation Mode

Changeover from injection to the recirculation mode of operation following a loss-of-coolant accident is accomplished automatically except for operation of the refueling water tank outlet valves which require operator action to close. We discussed the possibility with Combustion Engineering that administrative errors may result in leaving these valves open and lead to possible degradation of emergency core cooling system pump performance and loss of suction. Combustion Engineering has specified an interface requirement that locates the refueling water tank piping connection to the containment sump piping at an elevation with respect to the sump which would make the consequences of leaving refueling water tank isolation valves open acceptable.

It is Combustion Engineering's contention that with this interface, there is no need to close the refueling water tank isolation valves during the recirculation mode of operation to maintain the performance of the emergency core cooling system pumps. We conclude that this interface requirement provides the isolation needed to protect the emergency core cooling system pumps during the recirculation mode of operation and is acceptable. (See, also Section 6.3.2)

7.3.3 Actuation System Logic Power Supplies

There are two independent, selective two-out-of-four coincidence actuation system logics per engineered safety features system actuated. Each actuation logic serves one redundant group of engineered safety features equipment and is powered from four independent vital power supplies. Our review of the power supplies for the actuation logics revealed that the power supplies were interconnected in the actuation logic system, and that the redundant actuation logic systems were both powered from the same power sources. It was our concern that a single failure could compromise the independence of the vital power supplies and could result in loss of all engineered safety features actuation and protection functions.

712 286

We, therefore, required Combustion Engineering to modify the design so that the four independent vital power supplies will be connected in a manner so as not to negate the independence of the redundant engineered safety features actuation systems and the vital power supplies. Combustion Engineering modified the design in Amendme . 44 to the CESSAR in a manner which preserves the required independence, and is therefore acceptable.

7.3.4 Other Interface Requirements

Our review of the engineered safety feature systems revealed the following additional interface requirements that shall be satisfied in the balance of plant design:

(1) The power connections to the motor-operated valves located in the redundant high pressure hot leg injection lines shall be made to satisfy the single failure criterion. This requirement shall be satisfied both while providing the capability of achieving hot leg injection and while preventing the initiation of hot leg injection flow during the short term cooling period (established in the accident analysis) immediately following a loss-of-coolant accident.

(2) The power connections to the isolation valves in the safety injection and spray pump recirculation lines to the refueling water tank shall be made to satisfy the single failure criterion. This requirement shall be satisfied both while providing for recirculation flow when required and while preventing liquid from the containment sump to enter the refueling water tank during the recirculation mode of operation following a loss-of-coolant accident.

(3) Each source of auxiliary alternating current and standby onsite power shall be physically and electrically independent as required by General Design Criterion 17.

(4) The balance of plant design shall satisfy all the interface acceptance criteria listed for the engineered safety feature systems in Table 7-1 of this report.

7.3.5 Conclusions

We reviewed the engineered safety features actuation system as discussed above. Our review entailed the descriptive information which included functional logic diagrams, testing capabilities, control of bypasses, interface requirements, Combustion Engineering's proposed design criteria and design bases and Combustion Engineering's analysis of the adequacy of those criteria and bases. In applications referencing the CESSAR and having a balance of plant design for the engineered safety features actuation system that satisfy the interface requirements stated in the CESSAR and in Sections 7.3.1 and 7.3.4 above, there is reasonable assurance that the engineered safety features actuation system design proposed in the CESSAR will satisfy the Commission's requirements stated in Section 7.1 of this report.

On the basis of our review and the interface requirements specified in the CESSAR and in this report, we have concluded that the engineered safety features actuation system can meet the Commission's requirements and are acceptable.

7.4 Systems Required for Safe Shutdown

Systems identified in the CESSAR as being required for safe shutdown are: emergency power system, nuclear service water system, component cooling water system, emergency feedwater system, atmospheric dump system, shutdown cooling system, and chemical and volume control system (boron addition portion). The last two systems are within the System 80 design scope. The remaining systems will be evaluated during the review of user applications referencing System 80. Also, CESSAR identifies the instrumentation and control from System 80 that will be utilized, in conjunction with balance of plant design provisions (to be described in specific user applications) to place and keep the plant in a safe shutdown condition in the event that access to the main control room is restricted or lost.

The following sections address the problem areas revealed during our review and their resolutions.

7.4.1 Shutdown Cooling System

Two redundant and independent shutdown cooling system suction lines are utilized to remove residual heat from the core. Each line has three motor-operated valves arranged in series with two valves located inside and one outside the containment. Consistent with satisfying the requirements of General Design Criterion 34, the design of these valves must meet the single failure criterion. This requirement must be satisfied both while providing the capability of achieving cold shutdown from the control room and while preventing overpressurization of the shutdown cooling system. The instrumentation, control and electrical equipment pertaining to these valves must be designed to conform with IEEE Std 279-1971 and IEEE Std 308-1971.

Our review of the design revealed that the shutdown cooling system did not satisfy General Design Criterion 34, in the event of a single electrical failure in the shutdown cooling system suction line motor-operated valves. Also, the design did not satisfy the staff's postion with regard to achieving cold shutdown from the control room. Combustion Engineering amended the CESSAR with a design that conforms with our requirements and we conclude that the proposed design is acceptable.

7.4.2 Interface Requirements

Our review of the systems required for safe shutdown revealed the following additional interface requirements that shall be satisfied in the balance of plant design.

(1) The power connections to the shutdown cooling system suction line motor-operated valves shall be made to satisfy the single failure criterion. This requirement

shall be satisfied both while providing the capability of achieving cold shutdown from the control room and while preventing overpressurization of the shutdown cooling system.

(2) Provisions shall be made to accomplish residual heat removal through the atmospheric dump valves from the control room.

(3) The balance of plant design shall satisfy all the interface acceptance criteria listed for the systems required for safe shutdown in Table 7-1 of this report.

7.4.3 Conclusions

We reviewed the descriptive information pertaining to the systems required for safe shutdown including the interface design requirements for those balance of plant systems to be described in applications referencing CESSAR. Also, we reviewed the design features proposed in CESSAR for accomplishing safe shutdown of the plant from outside the main control room. The review included functional logic diagrams, interface requirements, Combustion Engineering's proposed design criteria and design bases, and Combustion Engineering's analysis of the adequacy of those criteria and bases. In specific applications referencing CESSAR System 80, with the balance of plant design for the systems required for safe shutdown satisfying the interface requirements stated in Section 7.4.2 above, there is reasonable assurance that the proposed designs for systems required for safe shutdown in CESSAR System 80 could be made to satisfy the Commission's requirements stated in Section 7.1 of this report.

On the basis of our review and the interface requirements specified in the CESSAR and this report we have concluded that the instrumentation and controls designs for the systems required for safe shutdown of the plant meet the Commission's requirements and are acceptable.

7.5 Safety-Related Display Instrumentation

Our review of the safety-related display instrumentation included the monitoring of the reactor trip system, engineered safety features and post-accident information. Our review of the information pertaining to the control element assembly position indication will be coordinated with review of the computer protection system (Section 7.2.1). As stated previously in the event that the computer protection system is found to be unacceptable, the St. Lucie Unit 1 backup design will be substituted. The St. Lucie design provides two independent control element assembly position indications. We conclude that this will satisfy our requirements and is acceptable. The design of the automatic bypass indication of a protective action at the system level is outside the design scope of System 80 and will be evaluated during the review of each user's application using the recommendations of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Plant Safety Systems" as the basis for evaluating the adequacy of the indication system.

We reviewed the design description, Combustion Engineering's proposed design criteria, Combustion Engineering's analysis of the adequacy of those criteria, and interface requirements. We further will review each user's application referencing CESSAR System 80 to determine that Regulatory Guide 1.47 has been satisfied as an interface requirement as well as the interface criteria listed for the post accident monitoring system in Table 7-1 of this report. Meeting these requirements provides reasonable assurance that the proposed safety-related display instrumentation in the CESSAR System 80 will satisfy the Commission's requirements stated in Section 7.1 of this report.

On the basis of our review and the interface requirements specified in the CESSAR and this report, we conclude that the safety-related display instrumentation can meet the Commission's requirements and is acceptable.

7.6 All Other Instrumentation Systems and Requirements for Safety

7.6.1 Environmental Qualification

Combustion Engineering has stated that all Class IE equipment in Combustion Engineering's scope will be qualified for use under specified environmental service conditions in accordance with IEEE Std 323-1974, without exception. Combustion Engineering will reference IEEE Std 323-1974 in every purchase specification for Class IE electrical equipment. Combustion Engineering has committed to participate in the orderly development of an industry wide qualification program under the sponsorship and direction of a recognized technical society or similar control body. In addition, Combustion Engineering has also documented that in specific instances where problems emerge when attempting to implement the aging requirements of IEEE Std 323-1974, one of the following methods, singularly or in combination, will validate the qualification for that equipment.

(1) Analyses based upon environmental tests

(2) Operating experience (taking into consideration in-service inspection and periodic tests, preventive maintenance)

(3) Type tests utilizing qualitative aging techniques (e.g., environmental cycling, operational cycling, elevated stress techniques, etc.)

From our review of the information documented by Combustion Engineering, we have concluded that the proposed criteria for the qualification of Class IE equipment in the CESSAR can facilitate development of a qualification program consistent with the objectives established in IEEE Std 323-1974 and that the above commitment provide an acceptable basis for the Preliminary Design Approval of Class IE equipment qualification program.

7.6.2 Separation and Identification of Safety-Related Equipment

Combustion Engineering has stated in the CESSAR that the separation and identification of safety-related components and systems will comply fully with the requirements of

IEEE Std 384-1974 as augmented by Regulatory Guide 1.75 "Physical Independence of Electric Systems." We reviewed the information presented in the CESSAR in support of this commitment and concluded that it is acceptable. However, the final acceptability of the separation for the proposed design would be predicated on the satisfactory resolution of the following two items at the final design stage.

(1) Our review of the design arrangements of the plant protection system cabinets revealed that they were interconnected through internal cable wireways. We discussed the possibility with Combustion Engineering that single events such as a fire in one cabinet could propagate to the other cabinets resulting in the loss ofunction. Combustion Engineering has agreed and documented in the CESSAR to either demonstrate the capability of the interconnection design between plant protection system cabinets to withstand all design basis events, or modify the design to assure complete independence of these plant protection system cabinets. The CESSAR states that the analysis and/or test results substantiating the adequacy of the interconnection design will be submitted for our review and approval at least two months prior to the fabrication of the equipment for installation in any plant that references the System 80 design. We consider this design commitment to be an acceptable resolution of this concern.

(2) The design provides for card mounted fusible links to be used in wiring between redundant Class IE channels for fault isolation. We discussed the possibility with Combustion Engineering about a failure of a single link to isolate a fault in a channel propagating to other channels which could result in the loss of function. Combustion Engineering has agreed and documented in the CESSAR to either demonstrate the capability of the fusible link design to isolate a faulted channel from the others under the worst case conditions, or modify the design to assure complete independence of Class IE channels. Also, it has been agreed and stated in the CESSAR that the analysis and/or test results substantiating the adequacy of the fusible links will be submitted for our review and approval at least two months prior to the fabrication of the equipment for installation in any plant that references System 80 Design. We consider this design commitment to be an acceptable resolution of this concern.

In applications referencing the CESSAR and having a balance of plant design that satisfies IEEE Std 384-1974 as augmented by Regulatory Guide 1.75 as interface design criteria, there is reasonable assurance that the proposed design criteria for the physical independence and identification of safety-related equipment in CESSAR System 80 will satisfy the Commission's requirements and is therefore acceptable.

7.6.3 Shutdown Cooling Overpressure Protection Interlocks

The proposed shutdown cooling system design provides three serially connected motor-operated valves in each shutdown cooling system suction line to isolate and protect the low design pressure shutdown cooling system from the high operating pressure of the reactor coolant system. Our review of the shutdown cooling system motor-operated suction valve interlocks revealed that the design did not satisfy Branch Technical Position EICSB-3 in Appendix 7A of the Commission's Standard Review Plan, with regard to providing protective interlocks to automatically close these valves when required. Combustion Engineering modified the design to provide for closing these valves automatically. In addition, Combustion Engineering has stated in the CESSAR, at our request, that on-line testing capabilities of the interlocks to prevent opening and automatic closing of these valves will be provided. It was also documented in the CESSAR the testability of the shutdown cooling system overpressure protection circuits will be the equivalent to that required for other engineering safety features circuits. We reviewed the information presented in the CESSAR in support of the design changes and concluded that the proposed design is acceptable.

We found the interlock design initially proposed in the CESSAR for protecting the low design pressure (400 pounds per square inch, gauge) shutdown cooling system from the high pressure (600 pounds per square inch, gauge) safety injection tank to be inadequate. The design did not meet the single failure criterion in that, following isolation of the safety injection tank and after opening the shutdown cooling system isolation valves to place the shutdown cooling system in operation, the opening of one of the safety injection tank isolation valves due to single failure in the electrical connections to the valves or operator error could rupture the pump seals in both coolant loops. To assure that a single failure cannot result in the loss of the shutdown cooling system, Combustion Engineering has proposed to (1) reduce the safety injection tank pressure from 600 to 400 pounds per square inch, gauge prior to opening the isolation valves between the shutdown cooling system and the reactor coolant system, (2) interlock the shutdown cooling system isolation valves so they cannot be opened until the reactor coolant system pressure is below the design pressure of the shutdown cooling system, and (3) provide two shutoff valves, in series, in the nitrogen supply line to each safety injection tank for the purpose of minimizing the possibility that valve leakage can repressure the safety injection tank after its pressure is reduced. In addition, Combustion Engineering has agreed and documented in the CESSAR that the two shutoff valves in the nitrogen supply line to each safety injection tank will be powered from independent buses and that the proposed design modifications will satisfy the requirements set forth in IEEE Std 279-1971 and IEEE Std 308-1971. We reviewed the information presented in the CESSAR in support of these commitments and concluded that the proposed design modifications, accomplished in accordance with the above mentioned standards, is an acceptable resolution of the concern of overpressurization of the shutdown cooling system by the safety injection tanks.

In applications referencing the CESSAR and having a balance of plant design for the shutdown cooling overpressure protection interlocks that satisfy the interface requirements specified in the CESSAR and in Section 7.4.2 and Table 7-1 of this report, there is reasonable assurance that the proposed interlock design in the CESSAR will satisfy the Commission's requirements, and is therefore acceptable.

712 292

7.6.4 Safety Injection Tank Isolation Valves

Each of the four safety injection tanks is provided with a motor-operated isolation valve which is manually closed during normal shutdown cooling operation of the reactor to prevent the contents of the safety injection tanks from being automatically discharged into the reactor coolant system. However, it is imperative that the four safety injection tanks isolation valves be open when the reactor coolant system is at pressure to afford the protection required in the event of an accident. We reviewed the design of the safety injection tanks valve circuits that assure automatic opening of these valves when required and that maintain the valves open when the reactor coolant system is at pressure. We conclude that the design satisfies Branch Technical Positions EICSB 4 and 18 in Appendix 7A of the Commission's Standard Review Plan, and is acceptable.

In applications referencing the CESSAR and having a balance-of-plant design for safety injection tank isolation valves that safisfy the interface requirements specified in the CESSAR and in Table 7-1 of this report, there is reasonable assurance that the proposed design for these valves can be made to satisfy the Commission's requirements, and is therefore acceptable.

We will review each balance of plant design to determine that the interface criteria listed for the safety injection tanks isolation valves in Table 7-1 of this report, are satisfied. Designs satisfying these requirements will comply with the Commission's requirements.

7.6.5 Safety Injection Tank Pressure Restoration

The System 80 design provides for the manual depressurization of the safety injection tanks to 400 pounds per square inch, gauge during plant cooldown and for manual repressurization of the tank to 600 pounds per square inch, gauge when the reactor coolant system pressure is being increased. We advised Combustion Engineering that the proposed administrative controls do not provide sufficient assurance that the safety injection tank pressure will be restored to that required by the safety analysis during the various modes of reactor operation. Combustion Engineering has subsequently documented in the CESSAR that the administrative controls will be supplemented with an audible alarm to alert the operator of low safety injection tank pressure when the reactor coolant system pressure reaches 700 pounds per square inch, gauge. This alarm will meet the single failure criterion and will be designed in accordance with the applicable Class IE requirements set forth in IEEE Std 279-1971 and IEEE Std 308-1971. We conclude that this is acceptable.

In specific applications, with the balance of plant design satisfying the applicable requirements of IEEE Std 279-1971 and IEEE Std 308-1971 for the low safety injection tank alarm, there is reasonable assurance that the repressurization of the safety injection tanks can be accomplished in an acceptable manner.

7.6.6 Channel Trip Input Bypass Status to Plant Computer

The design of the plant protection system (i.e., reactor trip system and engineered safety feature system) provides for an independent bypass for each trip input in each protective channel. In addition to indicating the bypasses at the plant protection system cabinets and control room boards, the status of each bypass is provided to the plant computer. In response to our concern about compromising the independence of the plant protection system as a result of a failure in the non-class IE plant computer, Combustion Engineering has agreed and documented in the CESSAR to either demonstrate the capability of the connection design between the plant computer and the plant protection system to withstand all design basis events without jeopardizing the independence of the plant protection system, or modify the design to assure complete independence of the plant protection system from the plant computer. The CESSAR states that the analysis and/or test results substantiating the adequacy of the connection design will be submitted for our review and approval at least two months prior to the fabrication of the equipment for installation in any plant that references the System 80 design. We consider this design commitment to be an acceptable resolution of this concern. We will review the implementation of this commitment during our review of the CESSAR System 80 Final Design Approval application.

7.6.7 Response Time Testing of the Plant Protection System

CESSAR states that the sensor response time check and the overall integrated response time test of each protection function from sensor to final actuated equipment is outside the scope of the System 80 design. At our request, however, Combustion Engineering has identified in the CESSAR methods that may be used for measuring the various safety analysis response times of the plant protection system. The selection of the methods to verify the response time of specific functions and other design provisions to facilitate response time tests of the plant protection system will be evaluated during the review of applications that reference the CESSAR System 80 design. Therefore, we consider the resolution to evaluate this aspect of the design during the review of the balance of plant in user's applications acceptable.

7.6.8 Containment High Pressure During Leak Test

Our initial review of the permissible bypass conditions for the plant protection system revealed that the design did not include the capability for bypassing the containment high pressure inputs during leak detection tests. Combustion Engineering subsequently modified the CESSAR to indicate that there is no permissible bypass condition for the containment high pressure trip inputs during leak detection tests and that those components which would be actuated during these tests will be administratively removed from service prior to the tests. The administrative procedures will be supplemented by the inoperable status indication system which will be designed in accordance with the recommendations of Regulatory Guide 1.47 as augmented by Branch Technical Position EICSB 21 presented in Appendix 7-A of the Standard Commission's Review Plan. We consider this commitment acceptable and will verify to implementation during our review of user's application that reference the CESSAR.

7.6.9 Conclusions

We reviewed the descriptive information pertaining to all other instrumentation systems and requirements for safety including their design bases and Combustion Engineering's analysis of the adequacy of those bases. In specific applications referencing CESSAR System 80, with the balance of plant design satisfying the interface requirements presented in the above subsections of Section 7.6, there is reasonable assurance that all other instrumentation systems and requirements for safety in CESSAR System 80 could be made to satisfy the Commission's requirements.

On the basis of our review and the interface requirements specified in the CESSAR and in this report, we conclude that all other instrumentation systems and requirements for safety meet the Commission's requirements and are therefore acceptable. Where commitments and interface requirements have been provided, we will review their implementation during our review of each user's application that references CESSAR.

7.7 Control Systems

The following control systems are identified in CESSAR as not required for safety: reactor control, reactor coolant system pressure control, pressurizer level control, feedwater control, steam bypass control, and boron control. The in-core instrumentation system, which includes both fixed and movable detectors, and the plant computer participation to sequence the movement of regulating control element assembly groups are also identified as not being required for safety. As discussed previously, the CESSAR has introduced a new computer-based core operating limit supervisory system which is utilized to assure that the operator maintains the reactor system within the conditions assumed by the safety analysis. This system was also identified by Combustion Engineering as not being required for safety.

With the exception of the core operating limit supervisory system and the control element assembly sequencing by the plant computer as it interacts with the core protection calculators, there were no major differences identified by Combustion Engineering in the instrumentation and controls for the above mentioned systems and those provided in previous Combustion Engineering designs.

The interactions between the plant computer, core operating limiting supervisory system and core protection systems (depicted on the computer protection system hardware configuration block diagram presented in Figure 7-1 of this report) will be considered as part of our generic review of the core protection systems (see Section 7.2.1). In the event that it is determined that the role of the plant computer and core operating limit supervisory system and interaction with the core protection calculators degrade the safety of the plant, it has been agreed and documented by Combustion Engineering that the equivalent to the core operating limit supervisory system and control element assembly sequencing by the plant computer functions as implemented in the St. Lucie Unit 1 facility will be used as the backup design. The staff has reviewed and found acceptable this aspect of the design in the St. Lucie Unit 1 facility. Since the implementation of the backup design in CESSAR System 80 insofar as it relates to the electrical, instrumentation and control aspects of the design can be readily accomplished without degrading the safety of the plant, we conclude that the commitment to establish the adequacy of the interconnections between the core protection calculators and core operating limit supervisory system and plant computer as part of the research and development program for the computer protection system, and provide a backup design is acceptable and satisfies our present evaluation requirements. The results of our generic review of this matter will be reported at the same time as indicated for the core protection calculators in Section 7.2.1 of this report.

We reviewed the design description presented in the CESSAR with regards to the aforementioned control systems. In specific applications referencing the CESSAR and having a balance of plant design that satisfies the interface criteria for the control systems in Table 7-1 of this report,there is reasonable assurance that the proposed design of the control systems in the CESSAR will satisfy the Commission's requirements stated in Section 7.1 of this report.

On the basis of our review and the interface requirements previously identified, we have concluded that failures in these control systems would not be expected to degrade the capability of the plant safety systems in any significant degree or lead to plant conditions more severe than those for which the safety systems are designed to protect against and that these control and instrumentation systems can meet the Commission's requirements, and are acceptable.

7.8 Anticipated Transients Without Scram

Our review of anticipated transients without scram is contained in Section 15 of this report.

8.0 ELECTRIC POWER

8.1 General

The offsite and onsite electric power systems are entirely outside the design scope of CESSAR System 80 and they will be described in each user's applications that reference the CESSAR. While it appears that most of the important interface information has been listed, the interface design criteria presented in the CESSAR required for assessing the adequacy of the overall System 80 design when it is mated with a balance of plant design was found to be incomplete. The interface information reflects the need of an electric power system that will have two redundant and independent division arrangments for alternating current power, and four redundant and independent division arrangements for direct current power. This is consistent with the required redundancy of safety related components and systems included in the 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 user's application which incorporates CESSAR to determine overall System 80 design conformance with the Commission's requirements. The following section supplements the interface information provided in the CESSAR that we have identified. The interface acceptance criteria are listed in Table 8-1 of this report for the offsite power system, onsite alternating current power system, and onsite direct current power system. It is our position that these interface requirements shall be fully satsified in the balance of plant electric power system design to validate the assumptions made in CESSAR System 80 accident analysis and to provide an acceptable basis for the staff's conclusion that the CESSAR System 80 design will satisfy the Commission's requirements.

8.2 Interface Requirements

The following additional interface requirements revealed during our review of the CESSAR System 80 electric power system design shall form the basis for our review of user's applications which utilize CESSAR:

(1) The current System 80 design does not provide for the disconnection of the reactor coolant pumps from the electric system in the event of an underfrequency decay rate condition. It is our concern that in the event of an underfrequency decay rate event, the motors remaining connected to their buses could slow down the pumps faster than was assumed in the accident analysis involving flow coastdown as a result of the pump's kinetic energy. In response to our concern, Combustion Engineering has included an interface requirement in the CESSAR of three Hertz per second for the limiting underfrequency decay rate. Combustion Engineering has

been advised that we require that the analysis substantiating the three Hertz per second under frequency decay rate be submitted for our review in the CESSAR System 80 Final Design Approval application.

For specific applications that reference CESSAR where credit is taken for reactor coolant pump coastdown, we will require that the applicant either demonstrate that the effects of electric grid disturbances on its plant are such that the limiting underfrequency decay rate of three Hertz per second is not exceeded, or provide reactor coolant pump breakers and associated instrumentation and controls that are designed and qualified in accordance with the requirements of IEEE Std 279-1971 and IEEE Std 308-1971 including that the reactor coolant pump breakers be located in a seismic Category I structure. This is consistent with Branch Technical Position EICSB 15 in Appendix 7-A of the Commission's Standard Review Plan.

(2) The balance of plant electric power system design shall satisfy all the interface acceptance criteria listed in Table 8-1 of this report for the offsite power system, onsite alternating current power system and onsite direct current power system.

8.3 Conclusions

We reviewed the interface requirements imposed on the electric power system by CESSAR System 80. In specific applications referencing CESSAR System 80, 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 stated in Section 8.2 above, there is reasonable assurance that the assumptions made in the CESSAR System 80 accident analysis with regard to safety system functions can be sustained and the proposed System 80 in totality could be made to satisfy the Commission's requirements.

TABLE 8-1
INTERFACE ACCEPTANCE CRITERIA FOR ELECTRIC POWER SYSTEM

CRITERIA	TITLE	OFFSITE POWER SYSTEM	ONSITE A.C. POWER SYSTEM	ONSITE D.C. POWER SYSTEM
10 CFR PART 50	CONTENTS OF APPLICATION			
	TECHNICAL INFORMATION	X	X	X
	TECHNICAL SPECIFICATIONS	X	X	X
	CODES AND STANDARDS	X	X	X
GENERAL DESIGN CRITERIA (GDC) APPENDIX A TO 10 CFR PART 50	(SEE STANDARD REVIEW PLAN TABLE 8-1 FOR SPECIFIC GDC & TITLE)	X	X	X
IEEE STANDARDS				
IEEE STD 279-1971			X	X
IEEE STD 308-1971		X	X	X
IEEE STD 317-1972			X	X
IEEE STD 323-1974			X	X
IEEE STD 334-1974			X	
IEEE STD 336-1974		X	X	X
IEEE STD 338-1971			X	X
IEEE STD 344-1975			X	X
IEEE STD 379-1972			X	X
IEEE STD 382-1972			X	X
IEEE STD 383-1974			X	X
IEEE STD 384-1974			X	X
IEEE STD 387-1972			X	
IEEE STD 450-1972				X
REGULATORY GUIDES (RGS)				
RG 1.6			X	X
RG 1.9			X	
RG 1.22		X	X	X
RG 1.29			X	X
RG 1.30		X	X	X
RG 1.32		X		X
RG 1.40			X	
RG 1.41		X	X	X
RG 1.47		X	X	X
RG 1.53			X	X
RG 1.62			X	
RG 1.63		X	X	X
RG 1.68		X	X	X
RG 1.70		X	X	X
RG 1.73			X	X
RG 1.75			X	X
RG 1.81			X	X
RG 1.89			X	X
RG 1.93		X	X	X
BRANCH TECHNICAL POSITIONS (BTPs)				
BTP EICSB 1			X	X
BTP EICSB 2			X	
BTP EICSB 6				X
BTP EICSB 7			X	X
BTP EICSB 8			X	
BTP EICSB 10			X	X
BTP EICSB 11		X		
BTP EICSB 17			X	
BTP EICSB 21		X	X	X
BTP EICSB 27			X	X

712 299

CESSAR

9.0 AUXILIARY SYSTEMS

The proposed auxiliary systems designs have been evaluated for their functional capability to meet the requirements of reactor safety and prevent radiological releases to the environment. In the course of our review, we have focused our attention on the design bases of the auxiliary systems including any safety related objectives of the systems and the manner in which these objectives are achieved. We have also focused our attention on the design interface requirements that must be met by the balance of plant designer or utility-user referencing the CESSAR in order to assure that the assumptions made in the evaluation of these system designs as described in the CESSAR are valid.

The proposed designs of the auxiliary systems are generally comparable in design and function to those reviewed and approved in other recent pressurized water reactor applications.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

The design of the new fuel storage racks and storage area will be supplied by applicants referencing the CESSAR in the balance of plant design; however, the CESSAR contains certain interface requirements for new fuel storage design.

We have, reviewed the interface requirements presented in the CESSAR and conclude that applicants referencing the CESSAR will have the necessary information to design the new fuel storage facility to meet the positions set forth in Regulatory Guide 1.13, "Fuel Storage Facility Design Basis," and the requirements of General Design Criterion 62. We, therefore, conclude that the interface information provided in the CESSAR is acceptable. We will review the implementation of these interface requirements for each application which utilizes the CESSAR.

9.1.2 Spent Fuel Storage

The design of the spent fuel storage racks and the spent fuel storage area will be supplied by applicants referencing the CESSAR in the balance of plant design; however, the CESSAR contains certain interface requirements for the spent fuel storage design.

We have, reviewed the interface requirements presented in the CESSAR and conclude that applicants referencing the CESSAR will have the necessary information to design the spent fuel storage facility to meet the positions set forth in Regulatory Guide 1.13, and the requirements of General Design Criterion 62. We, therefore, conclude that the interface information provided in the CESSAR is acceptable. We will review the implementation of these interface requirements for each application which utilizes the CESSAR.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The design of the spent fuel pool cooling and cleanup system will be supplied by applicants referencing the CESSAR in the balance of plant design; however, the CESSAR contains certain interface requirements for the spent fuel pool cooling and cleanup system design.

We have reviewed the interface requirements presented in the CESSAR and conclude that applicants referencing the CESSAR will have the necessary information to design the facility to meet the positions set forth in Regulatory Guide 1.29, and General Design Criterion 61. We, therefore, conclude that the interface information provided in the CESSAR is acceptable. We will review the implementation of these interface requirements for each application which utilizes the CESSAR.

9.1.4 Fuel Handling System

The fuel handling system will be an integrated system consisting of equipment, tools and procedures for refueling the reactor. The system will provide for handling and storage of fuel assemblies from receipt of new fuel to the shipping of spent fuel. The new fuel handling crane, spent fuel cask, overhead handling crane and facilities designs will be provided by the balance of plant designer; therefore, their acceptability will be evaluated in applications which reference the CESSAR.

The System 80 fuel handling system will include machinery and tools designed for underwater handling of spent fuel from the time it leaves the reactor until it is placed in a cask for shipment from the site. The fuel handling system design will comprise the basic reactor fuel handling machinery. Special tools for transfer of fuel from the reactor vessel to and through the fuel transfer tube into the spent fuel storage pit, spent fuel handling machinery, new fuel handling special tools and machinery and a dry sipping device for detecting cladding defects in irradiated fuel assemblies during refueling operations are part of the design.

In Amendments 40, 41 and 42 to the CESSAR (Section 9.1.4), Combustion Engineering has provided analyses of the consequences of dropping the reactor vessel head assembly onto the reactor vessel, the reactor vessel nozzles, and upper guide structure during fuel handling operations. Combustion Engineering stated that the orientation that resulted in the highest stresses on the vessel, supports, internals and nozzles is a "straight drop" where the head assembly comes to rest on the reactor vessel flange. We have reviewed the analysis and agree with this statement.

We have reviewed the results of their analysis and assumptions used for the analysis and conclude that the assumptions are conservative. Combustion Engineering stated that in the unlikely event of a reactor head assembly drop, the shutdown cooling supply flow path will remain functional.

We have reviewed the design bases, systems descriptions, systems operation, safety evaluation and interface requirements for the above items and determined that the proposed fuel handling system design will enable the balance of plant designer to provide a satisfactory facility design to permit safe handling of new and spent fuel. We, therefore, conclude that the System 80 fuel handling system design is acceptable. We will review the implementation of the fuel handling system design, criteria and interfaces for each application which utilizes the CESSAR.

9.2 Water Systems

9.2.1 Station Service Water System

The design of the station service water system will be supplied by applicants referencing the CESSAR in the balance of plant design. However, the CESSAR contains the heat loads, as design interface requirements, for the systems within the scope of the CESSAR.

We have reviewed the design heat loads presented in the CESSAR as interface requirements for the basic nuclear steam supply system for maximum conditions of operation (either normal or accident). Based on our review, we conclude that the information in the CESSAR will enable applicants referencing the CESSAR to design an acceptable station service water system. We, therefore, conclude that the heat load interface information provided in the CESSAR for the station water system is acceptable. We will review the station service water system for each application to ascertain that these heat load interfaces are appropriately considered.

9.2.2 Component Cooling Water System

The design of the component cooling water system will be supplied by applicants referencing the CESSAR in the balance of plant design; however, the CESSAR contains certain design interface requirements for the component cooling water system design.

We have reviewed the interface requirements presented in the CESSAR and conclude that applicants referencing the CESSAR have the necessary information to design the component cooling water system with respect to requirements imposed by the nuclear steam supply system. We, therefore, conclude that the interface information provided in the CESSAR is acceptable. We will review the implementation of these interface requirements for each application which utilizes the CESSAR.

9.2.3 Ultimate Heat Sink

While this system will be provided by applicants referencing the CESSAR as part of the balance of plant design, Combustion Engineering provided, as interface information, the maximum heat loads for the basic nuclear steam supply system. This information will enable applicants referencing the CESSAR to design the ultimate heat sink.

We conclude that the interface information provided in the CESSAR is acceptable. We will review the implementation of this requirement for each application which utilizes the CESSAR.

9.3 Process Auxiliaries

9.3.1 Chemical and Volume Control System

The proposed chemical and volume control system is designed to control and maintain reactor coolant inventory and also to control the boron concentration in the reactor coolant through the process of makeup and letdown. Purification of the letdown coolant by removal of corrosion and fission products will also be accomplished by the chemical and volume control system. Portions of this system will also supply high pressure injection of borated water into the reactor coolant system for emergency boration.

The system comprises a regenerative heat exchanger, letdown heat exchanger, purification filters, purification ion exchangers, deboration ion exchanger, volume control tank, charging pumps, boric acid batching tank, refueling water storage tank, holdup tank, reactor makeup water tank, boric acid makeup pumps, reactor water makeup pumps, holdup pumps, chemical addition tank, boric acid filter, reactor makeup filter, reactor drain pumps, reactor drain filter, reactor drain tank, equipment drain tank, preholdup ion exchanger, gas stripper package, boric acid concentrator package, boric acid condensate ion exchanger, seal injection filter, piping, valves and instrumentation. The instrumentation includes a radiation monitor (failed fuel detector) for continuous recording of the reactor coolant gross gamma and specific fission product gamma activity and a boronometer for continuous recording of the reactor coolant boron concentration.

The system requires a component cooling water supply to the letdown heat exchanger. The exchanger is sized to function with a component cooling water supply temperature of 105 degrees Fahrenheit. Those portions of the chemical and volume control system that are required for safe shutdown of the reactor will be designed to seismic Category I requirements.

We have reviewed the design bases, functional requirements, design criteria, interface requirements, system functions, system description, components and their classifications, system operation, design and safety evaluation, and other data included in the CESSAR, and conclude that the proposed chemical and volume control system is acceptable.

9.4 Air Conditioning, Heating, Cooling and Ventilation Systems

The design of the air conditioning, heating, cooling and ventilation systems required for both seismic Category I and non-seismic Category I facilities will be provided by applicants referencing the CESSAR as part of the balance of plant design; however, the CESSAR contains certain interface requirements for these systems.

We have reviewed the interface requirements presented in the CESSAR for the basic nuclear steam supply systems and conclude that applicants referencing the CESSAR have the necessary information to design these systems to accommodate the CESSAR System 80 design and that the information provided in the CESSAR is acceptable. We will review the implementation of these systems for each utility-user's application which utilizes the CESSAR.

712 304

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 General

The steam and power conversion system transforms the thermal output of the reactor into electrical power from a turbine-generator. The System 80 reference system design scope includes the steam generator. The balance of plant design will provide the remainder of the steam and power conversion system design. The interface points between the steam generator and the balance of plant design are at the steam generator feedwater and steam nozzles as indicated in CESSAR Figure 10.3-1.

CESSAR Sections 5.1.4, 7.3.3 and Table 10.3-1 describe interface requirements that are imposed by the CESSAR on the balance of plant design for the steam and power conversion system, including the turbine generator, main steam supply system, turbine bypass system, main feedwater and condensate system, and emergency feedwater systems. The interface requirements are imposed to assure that the System 80 reference system will perform as evaluated in the CESSAR.

We have reviewed these interface requirements and conclude that they adequately specify all balance of plant steam and power conversion system design requirements related to interfaces with the System 80 nuclear steam supply system design and are acceptable.

10.2 Turbine Generator

The design of the turbine generator and associated systems will be provided by applicants referencing the CESSAR in the balance of plant design; however, the CESSAR contains certain interface requirements pertaining to the turbine generator.

We have reviewed the interface requirements presented in the CESSAR for the turbine generator system and conclude that the information provided is adequate and acceptable. We will review the implementation of these interfaces for each application which references the CESSAR.

10.3 Main Steam Supply System

The design of the main steam supply system will be provided by applicants referencing the CESSAR in the balance of plant design; however, the CESSAR contains certain interface requirements pertaining to the main steam supply system.

We have reviewed the interface requirements presented in the CESSAR for the main steam supply system and conclude that the information provided is adequate and acceptable. We will review the implementation of these interfaces for each application which references the CESSAR.

10.4 Feedwater and Condensate System

The design of the feedwater and condensate system, including the auxiliary feedwater system (Combustion Engineering's emergency feedwater system) will be provided by applicants referencing the CESSAR in the balance of plant design; however, certain interface requirements pertaining to the feedwater and condensate system are provided in the CESSAR.

We have reviewed the interface requirements presented in the CESSAR pretaining to the feedwater and condensate system and conclude that they are adequate and acceptable. We will review the implementation of these interfaces for each application which references the CESSAR.

We are currently evaluating, on a generic basis, design and operating conditions that could result in damage to feedwater system piping as a consequence of feedwater flow instability. The results of this evaluation may result in further requirements being imposed upon the CESSAR to ensure that unacceptable damage will not result from feedwater hammer.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Source Terms

Radioactive materials may be released to the environment from the liquid waste processing system, gas waste processing system, the boron recycle system, 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 boron recycle system as a part of the chemical and volume control system, is within the standard scope of the CESSAR System 80 design.

The CESSAR does not include the radioactive waste management systems in its design scope. These systems will be provided in the balance of plant design. However, the CESSAR does include as interface information, the concentrations of radioactive materials in the primary coolant and the flow rates of streams that are input to the radioactive waste management systems. This interface information will be used as design bases for balance of plant designs. We will, therefore, perform a detailed evaluation of the radioactive waste management systems for any applications referencing the CESSAR to assure that the system designs and capacities will be adequate for meeting the demands of the facility.

11.2 Liquid Waste System

The boron recycle portion of the chemical and volume control system, described in Section 9.3.1 of this report, will be a potential release pathway for radioactive materials in liquid effluents. Although the system is designed to extensively recycle processed liquids, discharges of evaporator condensate will be required. Combustion Engineering considers that 100 percent of these liquids will be recycled for reuse in the plant, but in our analysis, we assumed that 10 percent of the treated waste will be discharged due to operational upsets and to control the tritium inventory in the plant. Spent demineralizer resins and evaporator concentrates from the boron recycle system will be periodically transferred to the solid waste management system for packaging and shipment offsite. The principal components that makeup the boron recycle system, along with their principal design criteria, are listed in Table 11.1.

The boron recycle system which is operated in the batch mode has a capacity of 29,000 gallons per day, whereas the expected continuous input rate to the system is 1700 gallons per day. The holdup tank storage capacity and the difference between the expected daily input rate and system design capacity provide adequate reserve for processing surge flows. Based on the above, we have concluded that the system design and capacity will be adequate to control releases of radioactive materials in liquid effluents from the boron recycle system during normal operation, including anticipated operational occurrences, in accordance with General Design Criterion 60 of Appendix A to 10 CFR Part 50.

Table 11.1

Design Parameters of Principal Components of
Boron Recycle System

Component	Number	Capacity (gallons per minute) Expected	Design
Pre-holdup Ion Exchanger	1	84	128
Boric Acid Concentrator	1	*20	20
Boric Acid Condensate Ion Exchanger	2	*20	100
Gas Stripper	1	84	140
Holdup Tank	1	--	450,000 gal

*The system is operated in the batch mode at the rate of 20 gallons per minute.

12.0 RADIATION PROTECTION

Section 12 of the CESSAR provides information to permit a determination that the proposed nuclear steam supply system has design features that contribute to maintaining radiation exposures to plant personnel operating and maintaining the system as low as practicable and within applicable limits. It describes design features for Combustion Engineering designed and supplied systems in which as low as practicable radiation exposure considerations are implemented. It describes how the operation, maintenance, arrangement and potential radiation levels must be considered in the design of the systems and equipment. It also describes how these considerations have been implemented for such high exposure systems as radwaste processing and handling systems. Equipment specifically discussed in this context in CESSAR Section 12 includes pumps, ion exchangers, filters, tanks, the concentrator package, the reactor vessel head vent, the reactor coolant system leakage control, refueling equipment, inservice inspection equipment, remote instrumentation, and equipment in the containment. Material and equipment selection is also discussed as it relates to maintaining radiation exposures as low as practicable.

Our review considered the specific details of all the items described as being design features for assuring that occupational radiation exposures will be as low as practicable. We found that the design features are consistent with the guidance of Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as Reasonably Achievable." We also found the design features consistent with the principles that designs should be implemented to (1) make equipment maintenance less necessary, (2) permit maintenance to be accomplished rapidly when it must be performed in a high radiation field, and (3) assure that, to the extent practical, most expected maintenance requirements will be able to be performed in reduced radiation fields. We find that Combustion Engineering has considered operations which result in significant radiation exposures to operating personnel, and has proposed design features that assure that the as low as practicable objective will be met.

The systems discussed have adequate design provisions to assure that occupational radiation exposures that relate specifically to design features of the nuclear steam supply system can be as low as practicable during normal operations, including refueling, maintenance and inservice inspection. It will be necessary that the user referencing the CESSAR develop an appropriate radiation protection program for the balance of plant design and for operation of the plant. This program will be described in the balance of plant portion of any application that references the CESSAR, and will be reviewed with such application.

We find the radiation protection provisions of CESSAR Section 12 to be consistent with the acceptance criteria specified in Regulatory Guide 8.8 which are based on operating experience regarding minimizing radiation exposure, and that these provisions are therefore acceptable.

12.1 Shielding

CESSAR Section 12.1, regarding shielding, refers to Section 5.1.4 which provides the interface requirement for the balance of plant designer that sufficient space be provided around the nuclear steam supply system for inservice inspection. This requirement relates to assuring that occupational radiation exposures will be as low as practicable. In addition, Section 12.1 provides quantitative (Curie) values for the radiation sources in the CESSAR reference system equipment. Maximum values are provided for 1.0 percent failed fuel as well as average values for 0.25 percent failed fuel. The reactor vessel sources are also provided, as gamma and neutron spectra at specified locations. In addition, dose rates for reactor coolant system components are provided, both for limiting expected operating conditions and for a plant shutdown time of 48 hours. These latter values have been based, in part, on the thickness of radioactive corrosion products and particulate matter suspended in the coolant that is deposited on the component surfaces during operation. This material is referred to as crud. Crud activities are characterized in the CESSAR as varying considerably due to various factors, but experience from operating reactors was used in evaluating the crud sources.

The reactor and process equipment source information is consistent with the requirements for calculating shielding parameters. We conclude that the source information is acceptable.

13.0 CONDUCT OF OPERATIONS

Information relating to the conduct of operations is not within the design scope of the CESSAR and will be provided in each application that references the CESSAR.

With regard to industrial security Combustion Engineering has stated that the design conservatisms that it provides for protection against a broad range of accidents also reduces the chances that an act of sabotage could result in jeopardizing the health and safety of the public. Furthermore, Combustion Engineering has stated that they will continue to review, as appropriate, the CESSAR design for changes that will provide additional protection against sabotage.

We will review the provisions for protection again t sabotage in applications that reference the CESSAR. We consider that the Combustion Engineering design for protection of the plant against acts of industrial sabotage are acceptable for the Preliminary Design Approval stage of the review process.

14.0 INITIAL TESTS AND OPERATION

CESSAR Section 14 describes a proposed test program for the System 80 reference system. The test program is divided into two major phases, preoperational testing and startup testing. The preoperational test phase is subdivided into individual system and/or subsystem preoperational tests, and integrated reactor coolant system heatup and preloading hot functional tests. The startup test phase is subdivided into initial core loading, postcore hot functional tests, initial criticality, low power physics tests, and power ascension tests. Combustion Engineering's anticipated role in relation to the utility-user, in the preparation, review, and execution of test procedures, as well as in the evaluation of the test results and in determining the need for changes to test procedures as may be required from the test results, is described. The Combustion Engineering site support organization and the qualifications of the personnel that could be involved with the test program are also described.

The proposed test program includes a description of the test objectives, prerequisites, and interfaces for each system and/or component test as it relates to the System 80 reference system. We have found that the development of the test program is consistent with the guidance set forth in Regulatory Guides 1.68 and 1.79, and the requirements of Appendix B of 10 CFR Part 50, and on this basis we conclude the startup and test plans proposed for the System 80 reference system are adequate to ensure that an acceptable program can and will be implemented.

15.0 ACCIDENT ANALYSIS

15.1 General

Combustion Engineering has performed safety analyses to evaluate the capability of the CESSAR System 80 plant to withstand normal and abnormal operational transients and a broad spectrum of postulated accidents without undue risk to the health and safety of the public. The postulated events have been classified by Combustion Engineering with respect to evaluation criteria as follows:

(1) Class I

(a) Does not induce fuel failures.
(b) Does not lead to a breach of containment barriers and fission product release.
(c) Does not require operation of any engineered safety features.
(d) Does not lead to significant offsite radiation exposures.

(2) Class II

(a) May induce fuel failures.
(b) May lead to a breach of barriers and fission product release.
(c) May require operation of engineered safety features.
(d) May result in offsite radiation exposures in excess of normal operational limits.

(3) Class III

(a) Very low occurrence probability.
(b) Provide information relevant to site acceptability and certain design and performance aspects of the plant.
(c) May require operation of engineered safety features.
(d) May result in significant offsite radiation doses within the limits of 10 CFR Part 100.

Combustion Engineering's classification of the events it has analyzed is itemized in Table 15.1 of this report.

712 313

TABLE 15.1

CATEGORIES OF TYPICAL TRANSIENTS AND FAULTS

Class I

Uncontrolled control element assembly withdrawal from a subcritical or low power condition, including control element assembly or temporary control device removal error during refueling.

Control element assembly misalignment.

Uncontrolled boron dilution.

Loss of forced reactor coolant flow.

Startup of an inactive reactor coolant loop.

Loss of external electrical load and/or turbine trip.

Loss of normal feedwater flow.

Loss of alternating current power to the station auxiliaries (station backout).

Excessive heat removal due to feedwater system malfunctions.

Excessive load increase.

Class II

Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, which actuate emergency core cooling.

Minor secondary system pipe break outside containment.

Inadvertent loading of a fuel assembly into an improper position.

Class III

Major secondary system pipe failure.

Major rupture of pipe containing reactor coolant up to and including double-end rupture of the largest pipe in the reactor coolant system (loss-of-coolant accident).

Waste gas decay tank rupture.

Steam generator tube rupture.

Control element assembly ejection accident.

Fuel handling accident.

Single reactor coolant pump shaft seizure.

15.2 Input Parameters for Safety Analyses

The trip set points and the assumed trip delay times used by Combustion Engineering in the CESSAR safety analyses are given in Table 15.2 of this report. The control element assembly drop time used, 3.0 seconds to reach the 90 percent insertion position, was based on previous measurements applicable to reactors with 14 x 14 fuel rod assemblies, manufactured by Combustion Engineering. Verification of the trip delay times will be accomplished during testing of the equipment. Rod drop times will be verified by Combustion Engineering as part of the 16 x 16 fuel test program discussed in Section 4.2.3 of this report.

The high local power density and low departure from nucleate boiling ratio trips provide the necessary overpower protection for anticipated transients. The high linear power level trip is used only in the control element assembly ejection accident analysis. Table 15.2 provides the trip set point values used in the analyses. The set points finally implemented will be required to be conservative with respect to the analysis set points, fully accounting for all sensor and process delays and uncertainties.

The System 80 reference system application is applicable to a plant which will be licensed to operate at a maximum core thermal power level of 3800 megawatts. For this core power level, the nuclear steam supply system thermal power is 3817 megawatts. Combustion Engineering has stated in CESSAR Section 1.2 that the licensing of this design to operate at any higher core power level will be requested only when the Commission issues notification, as stated in Regulatory Guide 1.49, that it will accept applications containing the higher power level and that, prior to such a request, all phases of the design will be fully analyzed and evaluated by Combustion Engineering for operation at that higher core power level.

CESSAR Section 1.1.2 also indicates that the System 80 reference system is designed to insure that there are more than adequate safety margins for operating at the licensed maximum core thermal power level of 3800 megawatts. This is done by analyzing and evaluating the design of safety related systems, and components, which prevent accidents or mitigate their consequences, at a core thermal power level of 4100 megawatts. Further, the accident analyses in CESSAR Section 15.4 which contribute to the demonstration of acceptability, are conservatively evaluated at 4100 megawatts, thermal. Combustion Engineering also stated that it has carefully assessed the methods used in the performance of these accident analyses at 4100 megawatts, thermal, and has confirmed that the results are conservative for operation at the licensed maximum core thermal power level of 3800 megawatts. In addition, the accident analyses contained in the application for final design approval will be evaluated at the licensed maximum core thermal power level of 3800 megawatts (provided that the Commission has not issued notification that it will accept applications for a higher maximum core power level).

We conclude that the CESSAR conforms with the intent of Regulatory Guide 1.49, and our evaluation of the Class III accidents, with the exception of the offsite radiological

TABLE 15.2

REACTOR PROTECTIVE SYSTEM TRIPS USED IN THE

CESSAR SAFETY ANALYSIS

	Analysis Set Point	Normal Set Point	Uncertainty	Trip Delay Time (seconds)
High Logarithmic Power Level, percent	2	1	+1 -0.5	0.4
High Linear Power Level, percent	129	125	±4	0.4
Low Departure from Nucleate Boiling Ratio	(1,2)	(1,2)	(1)	0.15
High Local Power Density	(3)	(3)	(3)	0.15
High Pressurizer Pressure, pounds per square inch, absolute	2422	2400	±22	0.9
Low Pressurizer Pressure, pounds per square inch, absolute	1500(4)	1600(4)	±5	0.9
Low Steam Generator Water Level, percent	5	10	±5	0.9
Low Steam Generator Pressure, pounds per square inch, absolute	848(5)	870(5)	±22	0.9

(1) The trip set point will be that value which assures a departure from nucleate boiling ratio equal to or greater than 1.3 (based on modified W-3 correlation) for steady state operating conditions and during the course of all anticipated transients.

(2) The low departure from nucleate boiling ratio trip incorporates a low pressurizer trip, nominally set at 1750 pounds per square inch, absolute, but assumed to be 1728 pounds per square inch, absolute in the safety analysis.

(3) The trip set point will be selected to prevent fuel centerline melting under steady state operating conditions and all anticipated transients.

(4) Indicated values are for normal operation. Below an operating pressure of 2000 pounds per square inch, absolute, a variable set point is provided to be manually reduced when decreasing pressure and which is automatically increased when increasing pressure.

(5) For accidents initiated at ultimate power level, this nominal set point is 800 pounds per square inch, absolute, corresponding to a set point of 778 pounds per square inch, absolute for the analysis.

712 316

consequences, is based on a core thermal power level of 3876 megawatts, even though the analyses covered a thermal power range up to 4100 megawatts.

Core physics parameters used in the accident analyses have been reviewed and found to be suitably conservative to represent the most adverse conditions of the core design throughout the first burnup cycle with respect to reactivity coefficients, control rod worths, and local power peaking factors, provided that operating configurations are restricted to considered patterns. If reload cores differ in any significant manner from the core described in the CESSAR and considered in our evaluation, they must be evaluated separately to ascertain that they cannot result in more severe transients than have been considered.

15.3 Technical Specification Limits Quantified by Accident Analyses

Results of the accident analyses are sensitive to the values of many operating parameters which define core conditions at the start of the transient and govern the response of the system model to the postulated accident condition. Technical specifications must specify operating parameter limits and trip set points such that there is no potential for transients of more severe consequences than those predicted for the postulated accidents that are evaluated in the CESSAR.

Limiting core operating conditions which have been quantified by the accident analyses and which depend on parameters that are monitored directly or indirectly by plant sensors are core power, maximum linear heat generation rate, minimum departure from nucleate boiling ratio, and reactor coolant average temperature. These parameters are related to Commission criteria for potential fuel damage and to the potential consequences of postulated accidents.

In addition, the axial and radial power distributions throughout the core are needed to evaluate the stored energy distribution and the fuel damage potential due to high centerline temperature or clad burnout. The power distribution is dependent on the configuration of the operating control element assemblies core physics, and thermal-hydraulic parameters related to core design and fuel loading and burnup distribution. Combustion Engineering proposes to maintain the linear heat generation rate and the departure from nucleate boiling ratio within limits by use of the core protection calculators (see Section 7.2.1) which will acquire and process the necessary sensor signals to continuously monitor the linear heat generation rate and the departure from nucleate boiling ratio. The computer protection system will trip the reactor whenever the maximum linear heat generation rate or minimum departure from nucleate boiling ratio exceed limits at the existing steady-state conditions or would exceed limits if any anticipated operational occurrences considered by Combustion Engineering were to initiate and continue at the projected rate from the existing conditions. The power distribution will be determined by measuring core average power, synthesizing the normalized core average axial power distribution from out-of-core neutron flux signals and control element assembly group positions, and by synthesizing the radial peaking

712 317

factors from control element assembly position measurements. The radial peaking factor synthesis will be dependent upon precalculated planar radial peaking factors as a function of control element assembly configuration, and will be adjusted to reflect the existing condition of azimuthal flux tilt magnitude.

As stated in Section 7.2.1, if the computer protection system does not prove to be acceptable, a backup system will be provided to perform the trip functions. The performance of this system is sufficient to control critical parameters to assure that accident consequences are consistent with Part 100 requirements. Specific computations demonstrating the adequacy of system performance in this regard will accompany any change to the backup system.

Combustion Engineering has proposed a second computer system, the core operating limit supervisory system (see Sections 4.3, 7.2.1 and 7.7), to assist in maintaining steady state core operating conditions within technical specification limits. The core operating limit supervisory system, based on data acquired for the normal plant core monitoring computer from an extensive in-core instrumentation system and from other process instrumentation, continually monitors core power distribution and computes core power operating limits based on peak linear heat rate and margin to departure from nucleate boiling; the margin to licensed power level is also monitored. The core operating limit supervisory system safety related functions include the following:

(1) Monitoring the azimuthal flux tilt magnitude; proposed technical specifications will require operation within the flux tilt allowance settings in the core protection calculators. This would be governed by the core operating limit supervisory system when it is in service and by periodic monitoring by the operator (with greater margin of uncertainty) when the core operating limit supervisory system is not in service.

(2) Enforcement of the technical specification limit on steady state peak linear heat rate will be based on the core operating limit supervisory system indicated margin when the core operating limit supervisory system is in service and would be based on the computer protection system margin to trip in the most limiting of the operable core protection calculator channels when the core operating limit supervisory system is not in service.

(3) Enforcement of the technical specification limit on the steady state departure from nucleate boiling ratio will be based on the core operating limit supervisory system indicated margin to the departure from nucleate boiling when the core operating limit supervisory system is in service and would be based on the computer protection system when the core operating limit supervisory systems is not in service.

(4) Operation within the licensed total core power level will be governed by the core operating limit supervisory system when the core operating limit supervisory system is in service and would be based on the highest core power calculated in the core protection calculators when the core operating limit supervisory system is not in service.

Values for steady state core operating limits governed by the core operating limit supervisory system output (or operator monitoring) have not been established by Combustion Engineering for the linear heat generation rate, departure from nucleate boiling ratio or the azimuthal tilt magnitude. Limits on peak linear heat rate will be required to be within the most limiting values used as initial conditions of the accident analyses (see Section 15.5.5).

We conclude that the proposed technical specification limits on steady state core operating conditions are adequate to assure that fuel damage limits are not exceeded for normal operation and anticipated transients. Tabulation of these limits including limit values quantified by the analyses are provided in Table 15.3. No limit has been proposed to assure that the stored energy in the reactor coolant will not exceed that evaluated in the loss-of-coolant analyses. We conclude that an additional technical specification limit must be imposed on core average temperature or an equivalent measure of the reactor coolant stored energy, not to exceed that used in the loss-of-coolant accident. Combustion Engineering is aware of our requirement and will propose a suitable limit at the Final Design Approval stage of our review.

Technical specifications limits as finally implemented will also be required to account for uncertainties to assure that the actual values for the parameters for which limits are indicated in Table 15.3 are not violated. They will also be required to include restrictions with regard to changes in core configuration, computer systems software for both the computer protection system and the core operating limit supervisory system, and computer program input constants which affect the capability of the computer systems to perform safety related design functions. Such changes will be reviewed by the staff.

15.4 Anticipated Transients

A number of plant transients can be expected to occur with moderate frequency as a result of equipment malfunctions or operator error in the course of refueling and power operation during the plant lifetime. Such transients meet the criteria of Class I events in the evaluation and classification presented by the applicant.

We compared the Class I events of Table 15.1 to typical anticipated events normally considered for safety review. The event involving an increase in reactor coolant inventory due to inadvertent operation of the emergency core cooling system or the chemical and volume control systems charging pumps is not included among the anticipated transients considered. This event is among those specified in Table 15.1 of the "Standard Format and Content of SARs for Nuclear Power Plants," Revision 1, October 1972. We have informed Combustion Engineering that an analysis of the event must be submitted in the CESSAR Final Design Approval application and be reviewed and approved before a Final Design Approval can be granted.

We have reviewed the analyses submitted for anticipated transients to ascertain that the transients do not violate the specific criteria which follow: 712 319

TABLE 15.3

OPERATION LIMITS FOR MONITORED CORE PARAMETERS

Parameter	Limit Value*	Safety Relevance
(1) Total Core Power	3876 megawatts, thermal (1721 megawatts, thermal for two pump operation)	(a) Stored energy in fuel (b) Stored energy in coolant (c) Fuel centerline temperature (d) Fuel clad integrity
(2) Maximum Linear Heat Generation Rate	12.5 kilowatts per foot, maximum	(a) Fuel centerline temperature
(3) Minimum Departure From Nucleate Boiling Ratio	1.3** minimum by Westinghouse Electric Corporation Modified W-3 Correlator	(a) Fuel clad integrity
(4) Reactor Coolant Average Temperature	Maximum value corresponds to 567.5 degrees Fahrenheit core inlet at 2876 megawatts, thermal	(a) Stored energy in coolant (b) Fuel centerline temperature (c) Fuel clad integrity (d) Stored energy in fuel
(5) Azimuthal Flux Tilt Magnitude	Not to exceed the value being used in core protection calculator calculation of power distribution	(a) Fuel centerline temperature (b) Fuel clad integrity

*The limit value is the value used in the CESSAR safety analysis; technical specifications must assure that measured values are less than the value in this table by sufficient margin to account for uncertainties.

**Reference Section 4.4.2.3 of CESSAR Safety Analysis Report. The steady state operating margin to departure from nucleate boiling is to be sufficient to prevent the indicated value from being exceeded for any anticipated operational occurrence.

712 320

(1) Pressure in the reactor coolant and main steam system should not exceed 110 percent of design pressure (Section III of ASME Boiler and Pressure Vessel Code).

(2) Clad integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio throughout the transient will satisfy the 95/95 criterion and that the maximum centerline temperature remains below the fuel melting point. The 95/95 criterion provides a 95 percent probability, at a 95 percent confidence level, that no fuel rod in the core experiences a departure from nucleate boiling.

(3) Other plant conditions of a more serious nature are not induced by the transient if other independent faults of a more serious nature have not occurred.

It was found that the most limiting transients in regard to core thermal margins were the loss-of-forced reactor coolant flow, part length control element assembly drop and single control element assembly withdrawal transients. For these transients, the minimum value of the departure from nucleate boiling ratio was approximately 1.3, which is also the limiting value accepted by the staff as evidence that clad integrity has not been jeopardized.

The most limiting transients with respect to pressure within the reactor coolant system were the loss-of-external electrical load transient and control element assembly withdrawal at 1 percent power. The peak reactor coolant system pressure of 2520 and 2530 pounds per square inch, absolute, respectively, did not result in violation of the 110 percent overpressure limits.

The boron dilution incident evaluation presented in Section 15.2.4 of CESSAR is consistent with the proposed technical specification limits for refueling boron concentration, and is acceptable.

We conclude that the plant design is acceptable with respect to transient response to events that might occur during the plant lifetime and that anticipated transients would not lead to more serious plant conditions in the absence of other faults.

15.5 Postulated Accidents

Combustion Engineering has analyzed the System 80 design to evaluate the effects and potential consequences of postulated accidents due to single faults which have small to extremely remote probability of occurrence. Such accidents meet the criteria of Class II and III events in the evaluation and classification presented by Combustion Engineering.

We have reviewed the accident analyses submitted by Combustion Engineering to assure completeness and conservatism in the analysis, and to evaluate the acceptability of the results.

We selected, for detailed analysis, five highly unlikely accidents that are representative of the spectrum of types and physical locations of postulated causes in the System 80 design and that involve the various engineered safety feature systems. The analyses of these accidents are discussed in the following paragraphs. The calculated effects on the core and the potential consequences of these accidents exceed or are expected to exceed those of all other postulated accidents that directly affect the System 80 design and are the same as those analyzed for previously licensed pressurized water reactor plants. The accidents analyzed were (1) control element assembly ejection, (2) reactor coolant pump motor seizure, (3) feedwater system piping breaks (4) steam piping breaks inside and outside of containment, and (5) reactor coolant system piping breaks.

15.5.1 Control Element Assembly Ejection

The mechanical failure of a control rod mechanism pressure housing could result in the ejection of a control element assembly. The consequences of this event 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, Combustion Engineering has analyzed the consequences of such an event.

The methods used to perform the analysis have been reviewed by the staff and found to be consistent with Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressuried Water Reactors." These include the use of the PDQ Code to determine radial peaking factors and a point kinetics representation of the core (CHIC-KIN Code) utilizing Doppler weighting factors calculated with the TWIGL Code.

The ejection analysis was performed for beginning-of-life and end-of-life conditions for both full power and zero power. To ensure a conservative analysis, the maximum radial peaking factor and the maximum ejected rod worth were assumed to occur for the same initial conditions, although calculations indicated that this is not necessarily true. Further conservatism was introduced by increasing both the maximum ejected control element assembly worth and radial peaking factor by 10 percent to account for uncertainties.

The results show that, in all cases analyzed, the enthalpy of the hottest pellet is below the 280 calories per gram limit specified in Regulatory Guide 1.77. The zero power cases produced the highest values, 274 and 265 calories per gram from the beginning-of-life and end-of-life conditions, respectively. Thus, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten uranium dioxide can be assumed not to occur. Combustion Engineering used 200 calories per gram as a clad damage threshold. At our request Combustion Engineering evaluated the clad damage based on the number of fuel pins experiencing a departure from nucleate boiling

TABLE 15.4

ASSUMPTIONS USED IN THE CALCULATION OF THE CONTROL ELEMENT ASSEMBLY EJECTION ACCIDENT DOSES

(1) Power is 4100 thermal megawatts

(2) Iodine decontamination factor of 10 between water and steam

(3) Primary and secondary coolant equilibrium concentrations as limited by technical specifications (1.0 microcurie of iodine-131 equivalent activity per gram and 100/$\bar{E}$ microcurie of noble gas activity per gram for primary coolant, and 0.1 microcurie of iodine-131 equivalent activity per gram for secondary coolant)

(4) Primary to secondary leak rate as limited by technical specifications (1 gallon per minute)

(5) Iodine source spike factor of 500 (rate of release from fuel) after accidents

(6) 1[illegible] percent of the iodine and noble gas activity in the fuel is in the fuel gaps

(7) All releases through the secondary system

(8) 7 percent of the fuel suffers clad failure after rod ejection accident

ratio of less than 1.3, the clad damage criterion given in Regulatory Guide 1.77. Results of the analysis based on this damage criterion show that events for beginning-of-life conditions produce clad damage in less than 1 percent of the fuel pins; results for events for end-of-life conditions indicate that clad damage will occur in less than 7 percent of the fuel pins.

Based on the conformance of the analysis with the recommendations of Regulatory Guide 1.77 we conclude that the analysis of the control element assembly ejection accident is acceptable. The radiological consequences of this accident are discussed in Section 15.5.6 of this report.

15.5.2 Reactor Coolant Pump Rotor Seizure

We reviewed Combustion Engineering's analysis of an instantaneous seizure of a rotor of a reactor coolant pump during any allowed mode of operation. This event was evaluated by Combustion Engineering, using computer codes (CESEC, TORC, and COAST) which are still under review by the staff. The mathematical model and parameters used as input were reviewed and found to be suitably conservative. The results of the analysis showed that less than 2 percent of the fuel rods experienced a departure from nucleate boiling. This assures that fuel damage will be minimal and that there will be no consequential loss of core cooling capability. The analysis showed that the maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressures.

Based on our review of the mathematical model, the input parameters and the analysis results, we have determined that, if the codes used in the analysis are approved by the staff, we will be able to conclude that the plant design is acceptable with regard to possible seizure of a rotor of a reactor coolant pump.

In the event that any of these three computer codes are found to be unacceptable, we will require this event to be reevaluated using an acceptable code and that the System 80 design be modified as required to assure that the results of a reactor coolant pump rotor seizure accident are acceptable to us.

We expect to complete our review of these codes on a schedule that will allow Combustion Engineering to revise the codes and modify the design, if necessary, and to submit the modifications in the CESSAR Final Design Approval application.

15.5.3 Feedwater System Piping Breaks

We reviewed Combustion Engineering analyses of a spectrum of feedwater line breaks inside and outside containment, during various modes of operation, and with or without offsite power. The accident which resulted in the most severe transient consequences was determined and evaluated using a mathematical model that is presently under review. The results of the analyses of maximum size double ended feedwater line break accidents between the steam generator and the feedwater line check valve showed that no fuel damage and no consequential loss of core cooling capability would result. The maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressures.

The results of Combustion Engineering's analyses as described in the CESSAR are acceptable; however, there are unresolved questions regarding the method used for these analyses. We requested Combustion Engineering to submit a topical report, describing, in detail, the feedwater line rupture accident analysis. Combustion Engineering has indicated in CESSAR Amendment 27 that it will issue this topical report in December, 1976. We expect to complete our review of this topical on a schedule that will allow Combustion Engineering to revise the analysis and modify the design if necessary, in the CESSAR Final Design Approval application.

15.5.4 Spectrum of Steam Piping Breaks Inside and Outside of Containment

We reviewed the analyses and effects of steam line break accidents inside and outside containment during various modes of operation and with and without offsite power. The accident which resulted in the most severe consequences was determined and evaluated by Combustion Engineering. The results of Combustion Engineering's analyses of the spectrum of steam line break accidents showed no expected fuel damage and no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio experienced by any fuel rod was shown to be greater than 1.3. The maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressures.

The analyses as described in the CESSAR indicate acceptable consequences, however some questions concerned with the analysis methods, e.g., flow mixing effects in the lower plenum, the moisture carry-over model, and three-dimensional reactivity feedback effects, are still under review. We have requested Combustion Engineering to submit a topical report describing in detail the assumptions and calculation techniques used for steam line break analyses. Combustion Engineering documented in CESSAR Amendment 27 that it will issue this topical in June, 1978. We expect to complete our generic review of this topical report on a schedule that will allow Combustion Engineering to revise the analysis and modify the design, if necessary, in the CESSAR Final Design Approval application.

In order to meet the steam line accident criteria the applicant has provided an interface requirement which specifies that steam line flow restrictors be provided in each steam line as close as practicable to the steam generator nozzles. We conclude that this interface requirement is acceptable.

15.5.5 Spectrum of Reactor Coolant System Piping Breaks

The emergency core cooling system criteria require that emergency core cooling performance be calculated in accordance with an acceptable evaluation model and for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered. In addition, the calculation is to be conducted with at least three values of the discharge coefficient applied to the postulated break area spanning the range from 0.6 to 1.0.

Amendment 41 to the CESSAR addresses a spectrum of nine breaks for the loss-of-coolant accident from major reactor coolant system pipe ruptures. Discharge coefficient values were considered which covered the required range of 0.6 to 1.0. Two analyses for hot leg and pump suction leg large breaks were included. Amendment 31 to the CESSAR included a spectrum of four small breaks.

We have reviewed the break spectrum provided in the CESSAR and conclude that the spectrum of reactor coolant system piping breaks is in conformance with 10 CFR Part 50, and therefore acceptable.

15.5.6 Radiological Consequences of Design Basis Accidents

We have evaluated the consequences of the design basis loss-of-coolant, steam generator tube rupture, main steam line break, and control element assembly ejection accidents and determined the limiting atmospheric diffusion factor required by a site in order to meet the guideline doses for site evaluation purposes. We used the guideline dose values of 150 rem to the thyroid and 20 rem to the whole body, as described in Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," to determine the acceptability of sites that are evaluated at the construction permit stage with respect to low probability design basis accidents.

The System 80 reference system does not include the containment in its design scope. For a hypothetical case where the containment design leak rate is 0.1 percent per day, the total containment volume is 3,000,000 cubic feet, of which a maximum of 20 percent is not accessible to the sprays, and for which a minimum mixing rate of 20,000 cubic feet per minute is established, a two-hour dose reduction factor of 6.3 can be achieved. With this dose reduction factor we calculate that a relative concentration of 4.7×10^{-4} seconds per cubic meter or less, is required for the CESSAR to meet the dose guidelines of Regulatory Guide 1.4 without additional engineered safety features for the design basis loss of coolant accident. The assumptions used in the analysis are given in Table 5.5.

Of those sites previously evaluated by the Commission, only approximately 30 percent had two hour relative concentration values equal to or less than 4.7×10^{-4} seconds per cubic meter at the exclusion area boundary. Sites with higher relative concentration values at the exclusion area boundary (i.e., poor atmospheric dispersion characteristics) will be required to meet dose guidelines by designing the spray additive system to achieve a higher dose reduction factor or by providing additional engineered safety features. Plants of recent design have adequate engineered safety features in the balance of plant to satisfy the dose guidelines.

TABLE 15.5

ASSUMPTIONS USED IN THE EVALUATION OF THE RADIOLOGICAL CONSEQUENCES OF THE LOSS OF COOLANT ACCIDENT

(1) Power level of 4100 megawatts thermal

(2) Operating time of three years

(3) Containment leak rate of 0.10 percent per day (0-24 hours)

(4) Iodine composition of 91 percent elemental, 4 percent organic, and 5 percent particulate

(5) A two-hour thyroid dose reduction factor for sprays of 6.3

Based on the information supplied in the CESSAR for the control element assembly ejection accident, we calculated that, for a 30 meter elevated release, a site with a two-hour relative concentration value of 1 x 10^{-3} seconds per cubic meter or less at the exclusion area boundary is required to meet the 150 rem thyroid dose guideline value. The assumptions used in the ejection accident are listed in Table 15.4. The radiological consequences are determined primarily by the amount of failed fuel and the assumed primary to secondary system leak rate. For sites with poorer dispersion conditions, a primary to secondary system steam generator tube leak rate below one gallon per minute may be required.

The resulting doses for the steam generator tube rupture and steam line break accidents are functions of the activity concentrations in the primary and secondary coolant as limited by the technical specifications. The coolant activities technical specification limits, however, were set previous to the advent of the new generation of large nuclear power plants (e.g., the System 80 design). We will, therefore, include appropriate limits on primary and secondary coolant activity concentrations in the technical specification for those plants utilizing the CESSAR System 80 design to maintain the doses within 10 CFR Part 100 guidelines even with iodine spiking.

15.6 Anticipated Transients Without Scram

Our generic evaluation of anticipated transients without scram has continued since issuance of WASH-1270 "Anticipated Transients Without Scram for Water-Cooled Power Reactors" in September 1973. While still not finished, our review is nearing completion. We have issued a status report on our review and have met with the Advisory Committee on Reactor Safeguards to discuss that report. We will factor the Committee's comments into the development of a final staff position on the matter. Our final conclusions as to the measures that must be taken to assure the acceptability of a specific plant design with respect to all the considerations associated with anticipated transients without scram must, of course, await the completion of our generic review. However, because of the staff's efforts and those of the nuclear steam supply system manufacturers during the past years, we and the manufacturers have a sound understanding of the type of calculational models that will be needed for the required assessments. Further, for certain designs, preliminary results obtained using such models and a range of values for certain dominant input parameters have provided an understanding of design measures that, if taken, would very likely be sufficient to eliminate the need for further design changes to make the consequences of an anticipated transient without scram acceptable on the basis of the finally approved calculational models and assumptions.

Combustion Engineering, Incorporated and the Commission staff both have sufficient understanding of the System 80 design and the models and assumptions that will likely result from the staff's ongoing generic review, to predict with assurance the types of design changes that may be necessary to assure the acceptability of the consequences of an anticipated transient without scram in a System 80 plant.

Based on the review performed to date, the following plant modifications are indicated:

(1) Additional pressurizer safety valve relieving capacity to meet the emergency stress intensity limit.

(2) Diverse means to automatically actuate systems such as the auxiliary feedwater system, opening of atmospheric dump valves and containment isolation must be provided.

(3) The logic associated with the auxiliary feedwater system must not interrupt the auxiliary feedwater flow due to low steam generator pressure.

(4) Diverse means must be provided for interrupting power to the rod drive mechanisms on reactor scram.

(5) The plant operating procedures for the auxiliary feedwater system valves must be revised to assure that the required auxiliary feedwater flow will reach the steam generators.

Based on our review to date these design changes should render the consequences of an anticipated transient without scram event acceptable in a System 80 plant. However, until the final results of an analysis by Combustion Engineering obtained through the use of approved final calculational models and assumptions are available we have concluded that it would be premature to require specific design changes to be made to the System 80 design. We intend to continue to review this matter and will require any changes indicated to be needed in the System 80 design by the result of approved analyses to be incorporated into the design in a timely manner. We will issue a Preliminary Design Approval for the System 80 design on this basis.

15.7 Conclusions

We have reviewed Chapter 15 (Accident Analyses) of the CESSAR and conclude that the analysis of abnormal transients and postulated accidents are acceptable for the purpose of Preliminary Design Approval of the System 80 reference system.

However, there are several matters that will be required to be completed subsequent to the Preliminary Design Approval, as discussed below, and have their resolution reported in the CESSAR Final Design Approval application.

(1) Technical specification limits on reactor coolant average temperature, or an equivalent measure of core stored energy must be established.

(2) Technical specification restrictions must be established to assure Commission review of changes in core configuration and computer systems software changes for both the core protection calculators and the core operating limit supervisory system which affect the capability of the computer systems to perform safety related functions.

(3) Control element assembly drop times used in the safety analyses must be verified by drop tests performed with the 16 x 16 fuel element assembly configuration.

(4) Trip delay times and uncertainties used to establish final trip set points within analyses values must be fully justified by test results.

(5) The expected range of reactor parameters (Doppler coefficient, moderator temperature coefficient, etc.) must be estimated for subsequent fuel cycles of each plant.

(6) The review of the CESEC and TORC codes must be completed.

(7) The generic review of the methods used for the analysis of the loss of flow transient, including the use of the COAST code must be completed.

(8) Reports on the steam line break and feedwater line break accidents must be submitted and reviewed and approved by the staff. Design modifications required as a result of the review must be submitted and reviewed and approved by the staff.

(9) An analysis of the transient associated with inadvertent operation of the emergency core cooling system or chemical and volume control system charging pumps must be submitted and reviewed and approved by the staff.

712 329

16.0 TECHNICAL SPECIFICATIONS

The technical specifications in an operating license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the Commission. Final technical specifications will be developed and evaluated at the final design review stage. However, in accordance with Appendix O, paragraph 3, of 10 CFR Part 50, an application for a preliminary design approval of a standard design is required to include preliminary technical specifications. The regulations require an identification and justification for the selection of those variables, conditions, or other items which are determined as a result of the preliminary safety analysis and evaluation to be probable subjects of technical specifications, with special attention given for those items which may significantly influence the final design.

We have reviewed the proposed technical specifications presented in Section 16 of the CESSAR in conjunction with our review of CESSAR Sections 1 through 15 with the objective of identifying those items that would require special attention at the preliminary design review stage, to preclude the necessity for any significant change in design to support the final technical specifications. The proposed technical specifications are similar to the standard technical specifications being developed for plants using nuclear steam supply systems designed by Combustion Engineering. Several of the technical specifications proposed in the CESSAR have been modified as a result of our review. The associated required design changes have been addressed in the applicable sections of this report wherein the particular aspect of the system design affected by the technical specifications is evaluated. We conclude that the revised preliminary technical specifications are acceptable at this review stage.

712 330

17.0 QUALITY ASSURANCE

17.1 General

Section 17 of the CESSAR describes the quality assurance program of Combustion Engineering's Power Systems Group for the design and procurement phases of the System 80 nuclear steam supply system. Our evaluation of the quality assurance program is based on a review of this information and discussions and meetings with Combustion Engineering to determine how its quality assurance program complies with the requirements of Appendix B of 10 CFR Part 50 and the applicable Regulatory Guides.

17.2 Organization

The group within the Combustion Engineering organization responsible for the design and procurement of the System 80 nuclear steam supply system is the Power Systems Group (Figure 17.1). The Group consists of two subGroups, General Services (Figure 17.2) and Nuclear Power Systems (Figure 17.3), each directed by a Vice President. Each Vice President reports to the President of the Power Systems Group.

The Director of General Services Quality Assurance has been delegated by the President of the Power Systems Group, through the Vice President, General Services, the responsibility for establishing and assuring implementation of the Power Systems Group quality assurance policies, goals, and objectives. The Director of General Services Quality Assurance determines that the mandatory requirements of policies, goals, and objectives are imposed on management by means of a quality assurance manual.

Combustion Engineering is required by the quality assurance manual to develop systems and procedures to implement the mandatory quality assurance policy. There are two organizations directly responsible for implementation of the quality assurance program, General Services Quality Assurance and Design Quality Assurance. The Director of General Services Quality Assurance, in addition to developing the quality assurance program, is responsible for supplier control, which includes control of Combustion Engineering's manufacturing organizations. The supplier control program includes (1) evaluation and approval of suppliers, (2) review and approval of procurement orders, (3) supplier surveillance and audit, and (4) review and approval of supplier's procedures, records, and certifications.

Design Quality Assurance is a section of Project Services within Nuclear Power Systems. The Manager of Design Quality Assurance is responsible for implementing quality policies in design to determine effectiveness of design control activities.

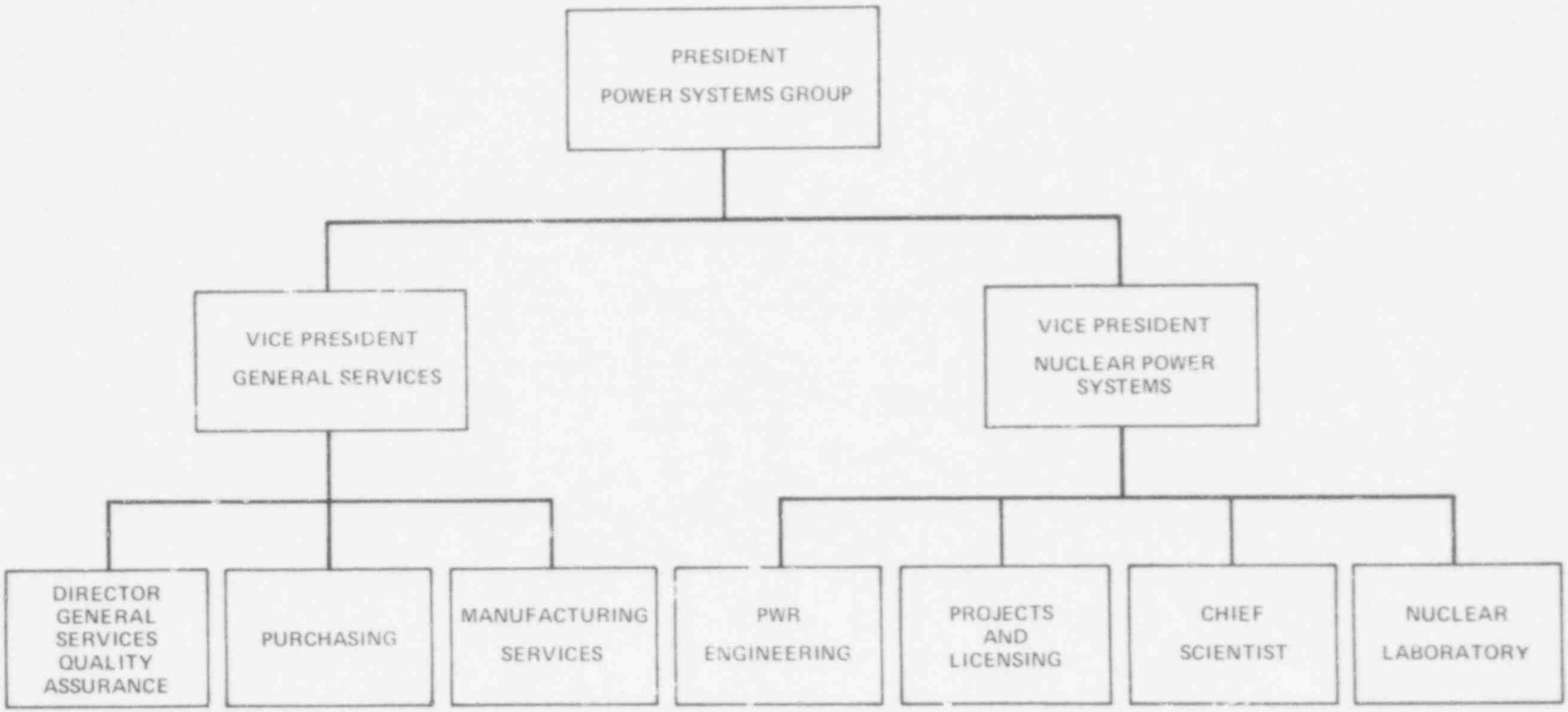


Figure 17.1 -- Power Systems Group

712 332

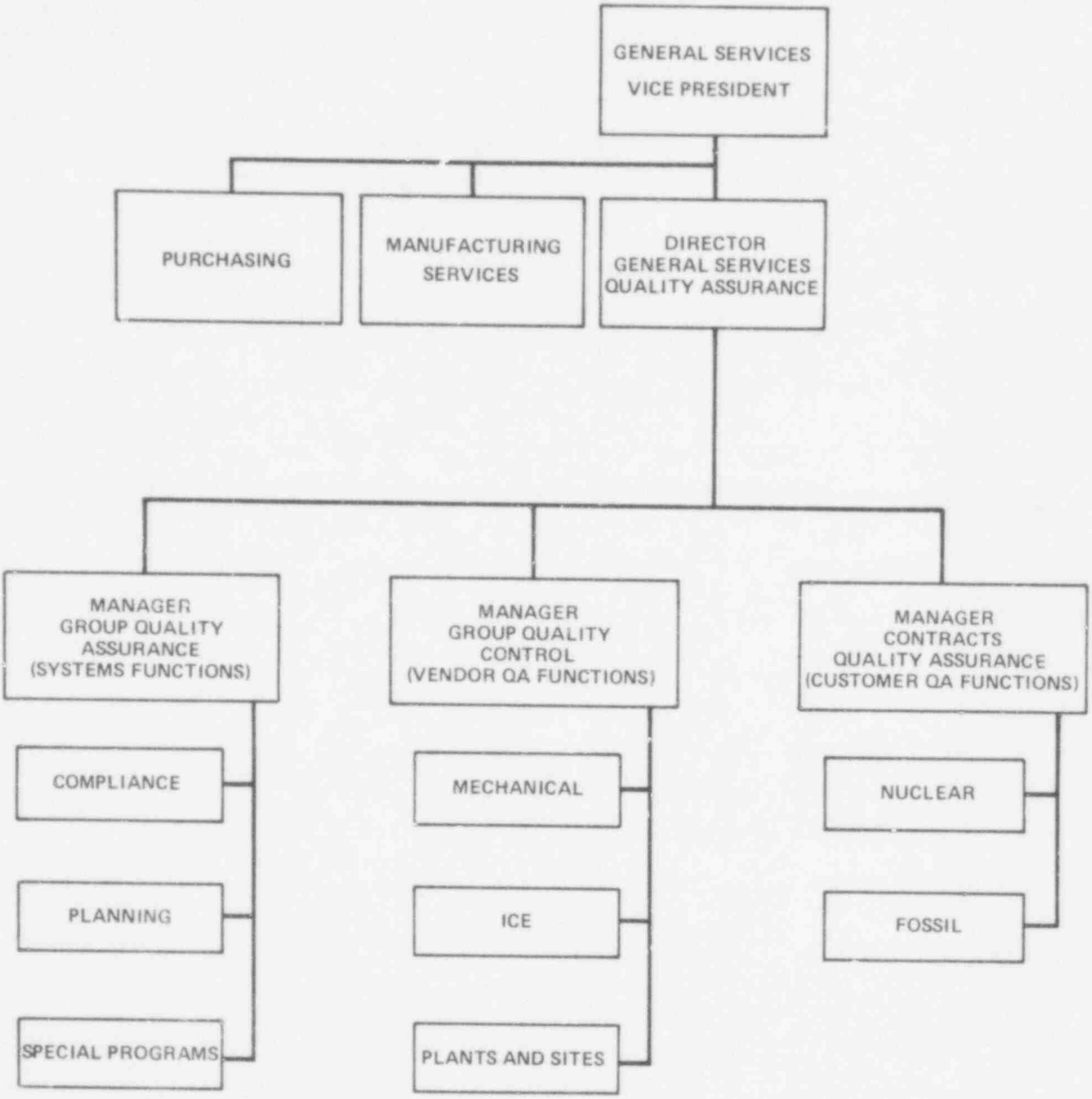


Figure 17.2 — General Services

712 333

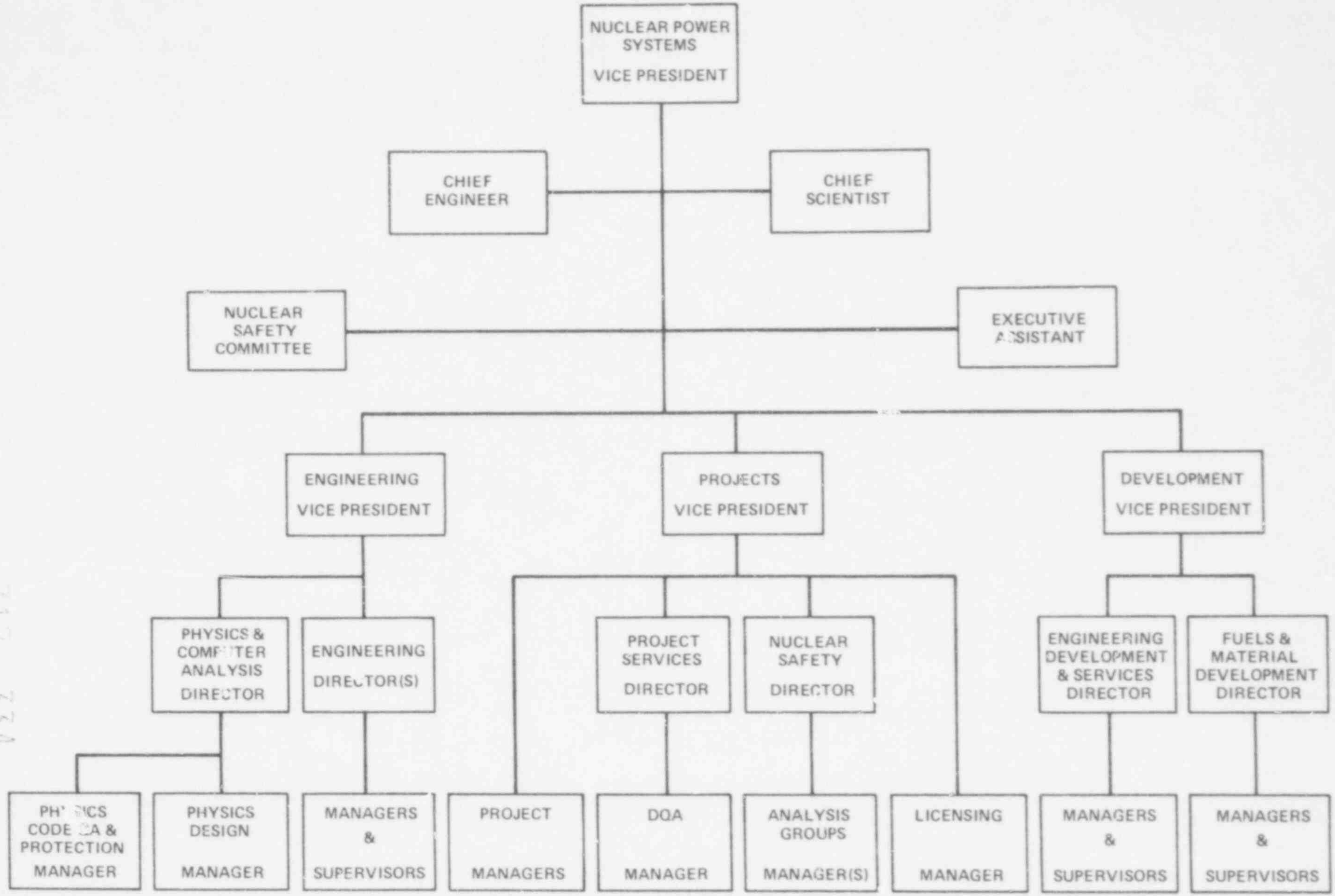


Figure 17.3 – Nuclear Power Systems

712 334

The right to stop work is delegated from the President to all levels of quality assurance management including General Services Quality Assurance and Design Quality Assurance. Disputes between personnel in quality assurance and personnel in engineering, purchasing, or supplier organizations are settled at the Director or Vice Presidential level when not resolved at lower levels.

Both quality assurance organizations have the authority and freedom to identify quality problems; to initiate, recommend, 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 condition has been approved.

17.3 Quality Assurance Program

The quality assurance program described in Section 17 of CESSAR applies to all safety-related items or services engineered and procured by Combustion Engineering, including nuclear fuel assemblies. Highlights of the quality assurance program are described in the following paragraphs.

Combustion Engineering maintains three levels of quality assurance on material. The first level, quality control, is conducted by the manufacturer's internal quality control organization. The second level, quality surveillance, is conducted by the General Services Quality Assurance personnel. These personnel conduct the third level of quality assurance by auditing first and second level activities.

The quality assurance program commits Combustion Engineering to meet the requirements of Appendix B of 10 CFR Part 50 and to follow the guidance provided by the Commission's Report WASH-1283, "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," dated June 7, 1973.

The quality assurance policies and procedures are documented in the Nuclear Quality Assurance Manual which includes "Methods and Procedures Instructions" and "Windsor Quality Assurance Documents." The CESSAR contains a brief description of these documents and shows their relationship to the 18 Criteria of Appendix B of 10 CFR Part 50.

Procedures require formal training and indoctrination of quality assurance personnel performing quality assurance related activities to assure competence in interpretation and implementation of the quality assurance manual.

The quality assurance program provides a design control system for structures, systems and components. 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. They also include measures to assure:

(1) That the design documents require compliance with Commission guidance and American National Standard Institute standards.

(2) That correct translation of Commission design bases and contractural requirements into specifications, drawings and procedures is performed.

(3) That the Manager of Design Quality Assurance systematically reviews and updates internal procedures to ensure effective implementation of the quality assurance program.

(4) That the Manager of Design Quality Assurance periodically performs audits to verify compliance with design quality elements and to assure that deviations are controlled.

(5) That project coordination and interface control preclude conflicting design objectives.

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. The review is performed by an individual who is independent of the original designer. Design Quality Assurance audits the process.

Combustion Engineering 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 Commission requirements, design bases, and quality requirements.

General Services 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 and verified by the Combustion Engineering Vendor Quality Assurance personnel.

Combustion Engineering requires that its suppliers identify and control materials, and Combustion Engineering inspects the marking of items prior to shipment. Material identification and control by suppliers is assured by requiring a written procedure which is reviewed by General Services Quality Assurance.

Combustion Engineering requires that in-process and final inspections be performed by suppliers in accordance with procedures submitted to and found acceptable by Combustion Engineering. Procedures require that supplier personnel be qualified and that records of qualification be maintained. These procedures require that supplier inspection personnel be organizationally independent from manufacturing personnel who perform the work being inspected.

Suppliers to Combustion Engineering must maintain a system providing for identification, documentation, and control of nonconforming items to prevent inadvertent use. General Services Quality Assurance reviews and approves nonconformance actions on "Deviation of Contract Requirements" forms. Engineering evaluates and dispositions nonconformances, and General Services Quality Assurance reviews these actions. General Services Quality Assurance also verifies proper corrective action of suppliers. Combustion Engineering provides nonconformance reports which are dispositioned "use as is" and "repair" to the user with the material.

Combustion Engineering executes a comprehensive system of planned and documented audits to verify product quality and compliance with the quality assurance program. The audits, with pre-established check lists, assure compliance with all aspects of Appendix B of 10 CFR Part 50, including the quality-related aspects of design, procurement, storage, shipment, and reactor site activities. Combustion Engineering's quality assurance program requires that suppliers also audit their operations and their subvendor's operations to verify conformance with quality requirements. The audits include quality related practices, procedures, instruction, and conformance with the quality assurance program. General Services Quality Assurance conducts audits of Combustion Engineering's manufacturing suppliers and their subvendors. Design Quality Assurance conducts biannual audits of the design control procedures of each functional engineering section to assure conformance with requirements. Written reports are forwarded to management of the area audited and to Combustion Engineering management. Follow-up audits assure corrective action.

17.4 Conclusions

We find that the quality assurance program described in Section 17 of the CESSAR provides for a comprehensive system of planned and systematic controls for design and procurement only. These controls adequately demonstrate compliance with each of the eighteen criteria of Appendix B of 10 CFR Part 50. We find that the quality assurance organizations have sufficient delegated independence and authority to effectively establish and execute the quality assurance program without undue influence from those directly responsible for costs and schedule.

We conclude that the quality assurance program for design and procurement as described in the CESSAR complies with the requirements of Appendix B of 10 CFR Part 50 and is acceptable.

Section 17 of the CESSAR does not describe quality assurance program for Combustion Engineering's manufacturing or site activities. This description will be supplied by each balance of plant designer or utility-user incorporating or referencing the CESSAR and will be reviewed for each individual license application.

712 337

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (the Committee) completed its review of the application for a Preliminary Design Approval for the CESSAR System 80 design at its 185th meeting held September 11-13, 1975. A copy of the Committee's report dated September 17, 1975, which contains certain comments and recommendations, is attached as Appendix C. The actions we have taken or plan to take in response to these comments and recommendations are described in the following paragraphs.

(1) The Committee recommended that, during the design, procurement, construction, and startup, timely and appropriate interdisciplinary system analyses be carried out to assure complete functional compatibility across each interface for the entire spectrum of anticipated operations and postulated design basis accident conditions.

We have transmitted the Committee's recommendations to Combustion Engineering for its consideration in proceeding with the CESSAR System 80 design. We have recognized the importance of defining the safety related interface information required to establish compatability of the CESSAR System 80 design with the balance of plant. As discussed in Section 1.7, we have concluded that this interface information is acceptable for the Preliminary Design Approval. However, as part of our long range effort to improve the implementation of the Commission's standardization policy, we have initiated a dialogue with the nuclear industry in an effort to develop improved procedures for defining interface requirements for standard plant designs. Through this effort and additional experience that will be gained in evaluating standard plant designs during the Final Design Approval stage, we will be able to assure functional compatability between the CESSAR System 80 design and the balance of plant design.

(2) The Committee stated that the Nuclear Regulatory Commission staff has identified a number of outstanding issues specific to the CESSAR application which will require resolution before the issuance of a Preliminary Design Approval. The Committee recommended that these matters be resolved in a manner satisfactory to the staff.

Prior to the completion of the Committee's review of the CESSAR System 80 design, we had advised the Committee of a number of outstanding issues, which are identified in our Report to the Advisory Committee on Reactor Safeguards for the CESSAR. Both Combustion Engineering and the staff discussed these matters with the Committee at its 185th meeting. The satisfactory resolution of these outstanding issues is discussed in the appropriate sections of this report. We have resolved all of the outstanding issues specific to the CESSAR application in a manner acceptable to the staff.

712 338

(3) The Committee also requested that they be kept informed on the resolution of the following issues:

(a) The emergency core cooling system evaluation.

Our evaluation of the emergency core cooling system is addressed in Section 6.3 of this report. This matter has been resolved in a manner satisfactory to the staff. By issuance of this report, the Committee is being informed of the results of our evaluation.

(b) The analysis of the effects of anticipated transients without scram.

Our evaluation regarding anticipated transients without scram is addressed in Section 15.6 of this report. This matter is under generic review by the staff. We will report final resolution of this issue to the Committee.

(c) Generic review of the effects of failures of reactor pump lubricating oil and component cooling water systems.

Our evaluation of the reactor coolant pump lubricating oil and component cooling water systems is addressed in Section 3.2.1 of this report. This matter has been resolved in a manner satisfactory to the staff. By issuance of this report the Committee is being informed of the results of our evaluation.

(4) The Committee expressed its continuing concern regarding generic problems related to light water reactors, recommending that such problems be dealt with appropriately by the applicants and the Nuclear Regulatory Commission staff. These generic problems are discussed in a report by the Advisory Committee on Reactor Safeguards dated March 12, 1975. These generic problems are being worked on by the staff, various reactor vendors and other industrial organizations and will be the subject of continuing attention by the staff.

(5) The Committee stated that it needs to be assured of the dependability of in-core neutron flux sensors for control of reactors operating at low core power peaking factors, and recommended that the Commission staff and Combustion Engineering continue to gather pertinent information from operating Combustion Engineering reactors.

We will continue to gather and review such pertinent data and report to the Commmittee relevant information which tends to confirm the dependability of these sensors. Also, we will require that appropriate technical specifications be provided to assure that a sufficient complement of incore detectors is available to monitor core power distributions.

712 339

(6) The Committee encouraged Combustion Engineering and the Commission staff to accelerate their efforts towards developing computational methods to provide best estimate analyses of loss-of-coolant accidents and other postulated accidents.

On October 20, 1975, 1975, Combustion Engineering presented to the Committee's subcommittee for the Palo Verde Nuclear Generating Station (Docket Nos. STN 50-528, 50-529 and 50-530) the results of Phase 1 of their best estimate analysis program. We will keep the Committee informed of new developments pertaining to the development of computational methods for best estimate analyses of loss-of-coolant accidents and other postulated accidents.

(7) The Committee stated that the CESSAR design should include provisions which anticipate the maintenance, inspection, and operation needs of the plant throughout its service life, including cleaning and decontamination of the primary coolant system, and eventual decommissioning. In particular, the Committee believes that the Commission staff and Combustion Engineering should review methods and procedures for removing accumulations of radioactive contamination whereby maintenance and inspection programs can be more effectively and safely carried out.

During the past year, the staff has been reviewing the activation product problem, including data on occupational radiation exposures related to activation products and methods and procedures for preventing or reducing and removing accumulations of radioactive contamination in the primary coolant system of light water reactors. Information gathered at conferences on decontamination and decommissioning and inputs from specific technical sources in industry have resulted in the staff examining this issue in more detail in the review of nuclear power plant applications.

The draft working paper for Regulatory Guide 8.8, Revisions 2, "Information Relevant to Assuring that Occupational Radiation Exposure at Nuclear Power Stations will be As Low as Reasonably Achievable," incorporated all of the staff's findings to date on methods and procedures that are effective in reducing radioactive exposure related to activation products, and was reviewed by industry in April 1975. Two areas we have identified where information is lacking or insufficient are (1) the costs associated with various measures, and (2) the quantitative benefits in reduction of occupational radiation exposure associated with the measures. Because we do not have this information, we do not require additional design features for exposure reduction in plants presently under review.

We are continuing our study of the problems associated with decommissioning but, as yet we do not require specific design provisions for decommissioning. A few reactors have been decommissioned and we know from this experience that the resultant exposures can be kept within acceptable bounds. Because the experience in this area has been acceptable to date, we plan to continue our investigations further into this matter before we require that any special features be incorporated during the design and construction of a plant.

(8) The Committee stated that they believe that Combustion Engineering and the staff should continue to review CESSAR for design changes that will further improve protection against sabotage.

The Office of Nuclear Regulatory Research has funded studies concerning possible modes of sabotage at nuclear power plants. Any recommendations resulting from these studies, regarding additional design features to protect against acts of industrial sabotage, will be considered by the staff for incorporation in the System 80 design.

(9) The Committee recommended that the Commission staff perform an independent check on the calculation of steam generator blowdown force effects for postulated ruptures of the steam generator feed line.

We plan to perform independent analyses of steam generator blowdown force effects for postulated ruptures of the steam generator feed line. We will inform the Committee of the results of our analyses.

(10) The Committee recommended that for a standard reactor of this size, larger safety margins, such as obtainable from higher reflooding rates, should be demonstrated. Programs underway by Combustion Engineering include analytical and experimental studies aimed at providing the technical base for emergency core cooling system model improvements, as well as studying possible changes involving augmented emergency core cooling systems. The Committee believes that the programs constitute a sufficient basis for proceeding at this time, and that the demonstration of larger safety margins should be part of the first major revised version of the Reference System 80 design.

Studies are being conducted by the staff and several reactor vendors to define better the current safety margins associated with emergency core cooling systems. We will utilize the results of these studies in evaluating any future proposed modifications to the CESSAR System 80 emergency core cooling system.

712 341

19.0 CONCLUSIONS

Based on our analysis of the proposed CESSAR System 80 nuclear steam supply system preliminary design, we have determined that, subject to the conditions discussed herein, we conclude that for the portion of the nuclear reactor design covered by CESSAR System 80:

1. Combustion Engineering has described, analyzed and evaluated the proposed design including, but not limited to, the principal engineering criteria for the design; the interface information necessary to assure compatibility between the submitted design and the balance of nuclear power plant; the envelope of site parameters postulated for the design; the quality assurance program to be applied to the design, procurement and fabrication of safety related features of the nuclear steam supply system; the design features that affect plans for coping with emergencies in the operation of the reactor or major portion thereof; and has identified the major features and components incorporated therein for the protection of the health and safety of the public.

2. Such further technical or design information as may be required to complete the safety analysis will be supplied prior to or in the final design application.

3. Safety features or components which require research and development have been identified by the Combustion Engineering and it has described, and will conduct, research and development programs reasonably designed to resolve safety questions associated with such features or components.

4. On the basis of the foregoing, there is reasonable assurance that: (i) such safety questions will be satisfactorily resolved at or before the issuance of the operating license for the first nuclear power plant utilizing the CESSAR System 80 nuclear steam supply system; and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, a facility can be constructed and operated without undue risk to the health and safety of the public, provided the site characteristic conform to the distance requirements of 10 CFR Part 100, and provided further that the balance of plant is properly designed and constructed in conformity with the interface requirements specified in the CESSAR and in this report, as discussed above.

5. Combustion Engineering is technically qualified to design the nuclear steam supply system described in the CESSAR System 80 document.

712 342

APPENDIX A

CHRONOLOGY OF RADIOLOGICAL REVIEW

OF

COMBUSTION ENGINEERING STANDARD SAFETY ANALYSIS REPORT (CESSAR)

September 17, 1973	Application tendered for acceptance review.
September 25, 1973	Letter to Combustion Engineering regarding receipt of tendered application.
October 10, 1973	Meeting with Combustion Engineering to discuss correction of deficiencies in CESSAR draft.
October 25, 1973	Letter to Combustion Engineering rejecting tendered application and requesting additional information.
November 2, 1973	Meeting with Combustion Engineering to discuss the staff's acceptance review of CESSAR.
November 3, 1973	Letter to Combustion Engineering regarding anticipated transients without scram.
November 7, 1973	Letter from Combustion Engineering regarding proposed schedule for submission of additional information.
November 8, 1973	Letter to Combustion Engineering regarding anticipated transients without scram.
December 7, 1973	Submittal of revised CESSAR copies for a second acceptance review.
December 12, 1973	Letter to Combustion Engineering advising that CESSAR is sufficiently complete for docketing.
December 17, 1973	Response from Combustion Engineering regarding the staff's November 8, 1973 letter concerning anticipated transients without scram.
December 18, 1973	Letter from Combustion Engineering resubmitting CESSAR for detailed review.
December 19, 1973	CESSAR docketed.
December 19, 1973	Letter from Combustion Engineering providing schedule for submittal of additional information requested in the staff's October 25, 1973 letter.
December 26, 1973	Notice of receipt of Standard Safety Analysis Report.
December 26, 1973	Letter to Combustion Engineering informing them that CESSAR is docketed.
January 4, 1974	Notice of receipt of CESSAR published in FEDERAL REGISTER (39F.R.1090).
January 8, 1974	Meeting with Combustion Engineering to discuss interpretation and Combustion Engineering's response to our request for additional information.

712 343

January 17, 1974	Submittal of Amendment 1 to CESSAR consisting of partial responses to the staff's October 25, 1975 request for additional information.
February 14, 1974	Submittal of Amendment 2 to CESSAR consisting of partial responses to the staff's October 25, 1975 request for additional information.
February 15, 1974	Letter to Combustion Engineering in response to its December 17, 1973 letter on anticipated transients without scram.
March 4, 1974	Meeting with Combustion Engineering to discuss additional information regarding its quality assurance program.
March 6, 1974	Meeting with Combustion Engineering to discuss draft questions related to staff's review of CESSAR.
March 13, 1974	Letter from Combustion Engineering requesting that System 80 customers be placed on automatic distribution of CESSAR correspondence.
March 21, 1974	Meeting with Combustion Engineering to discuss draft questions related to staff's review of the quality assurance program described in Section 17 of CESSAR.
March 22, 1974	Letter from Combustion Engineering concerning basis for its request that the CESSAR proprietary appendix be withheld.
March 22, 1974	Letter to Combustion Engineering requesting additional information required as a result of the staff's review of CESSAR.
March 26, 1974	Letter to Combustion Engineering transmitting a review schedule.
March 26, 1974	Meeting with Combustion Engineering to discuss the staff's requests for additional information.
March 29, 1974	Letter to Combustion Engineering requesting additional information and enclosing a description of an acceptable seismic qualification program for electrical and mechanical equipment, plus a staff position on the application of the single failure criterion to manually-controlled electrically-operated valves.
April 2, 1974	Meeting with Combustion Engineering to discuss the staff's request for additional information.
April 4, 1974	Letter to Combustion Engineering requesting additional information required as a result of the staff's review of CESSAR.
April 9, 1974	Letter from Combustion Engineering regarding its schedule for responding to our March 22, March 29, and April 4, 1975 requests for additional information.
April 12, 1974	Letter to Combustion Engineering granting a request that System 80 customers be placed on automatic distribution of CESSAR correspondence.
May 10, 1974	Letter from Combustion Engineering regarding the schedule for responding to our request for additional information.
May 14, 1974	Letter from Combustion Engineering providing justification for withholding proprietary appendix to CESSAR.
June 2, 1974	Meeting with Combustion Engineering to discuss quality assurance programs.

712 344

June 3, 1974	Submittal of Amendment 3 to CESSAR consisting of partial responses to the staff's March 22, March 29, and April 4, 1975 requests for additional information.
June 10, 1974	Letter to Combustion Engineering granting withholding of Proprietary Volume 5 to CESSAR.
June 12, 1974	Meeting with Combustion Engineering to discuss cutoff dates for applicability of new Regulatory Guides to CESSAR.
June 27, 1974	Letter to Combustion Engineering regarding cutoff dates for applicability of new Regulatory Guides to CESSAR.
July 1, 1974	Meeting with Combustion Engineering to discuss the quality assurance program described in Amendment 3 to CESSAR.
July 1, 1974	Submittal of Amendment 4 to CESSAR consisting of partial responses to the staff's March 22, March 29, and April 4, 1975 requests for additional information.
July 12, 1974	Letter from Combustion Engineering providing draft responses to some of the staff's requests for additional information.
July 17, 1974	Letter from Combustion Engineering providing draft responses to some of the staff's requests for additional information.
July 18, 1974	Letter to Combustion Engineering requesting additional information required as a result of the staff's review of CESSAR.
July 21, 1974	Meeting with Combustion Engineering to discuss quality assurance questions.
July 22, 1974	Letter to Combustion Engineering transmitting a revised review schedule.
July 24, 1974	Meeting with Combustion Engineering to discuss its response to a question concerning energy release to the containment in a steam line break accident.
July 30, 1974	Meeting with Combustion Engineering to discuss the use of optional systems in the CESSAR.
August 5, 1974	Letter from Combustion Engineering providing a schedule for submittal of additional information.
August 8, 1974	Submittal of Amendment 5 to CESSAR consisting of responses to the staff's comments on the Combustion Engineering quality assurance program.
August 9, 1974	Meeting with Combustion Engineering to discuss the technical details of responses to Section 15 requests for additional information.
August 14, 1974	Letter to Combustion Engineering requesting additional information required as a result of the staff's review of CESSAR.
August 19, 1974	Submittal of Amendment 6 to CESSAR consisting of partial responses to the staff's March 22, March 29, April 4, and July 18, 1974 requests for additional information.
August 22, 1974	Letter to Combustion Engineering requesting additional information required as a result of the staff's review of CESSAR.
August 22, 1974	Meeting with Combustion Engineering to discuss the Auxiliary and Power Conversion System Branch's request for tabulation of system scope and interface requirements.

August 28, 1974	Meeting with Combustion Engineering to discuss additional information to be provided in response to staff's Section 15 requests for additional information.
August 30, 1974	Letter to Combustion Engineering requesting additional information concerning the computerized protection system.
September 3, 1974	Letter to Combustion Engineering requesting additional information required as a result of the staff's review of CESSAR.
September 4, 1974	Meeting with Combustion Engineering to discuss the treatment of exceptions to CESSAR being taken by Arizona's Palo Verde application and the procedure Combustion Engineering recommends for handling such exceptions on CESSAR plants.
September 5, 1974	Letter from Combustion Engineering providing a schedule for responses to staff's requests for information.
September 6, 1974	Meeting with Combustion Engineering to discuss some of the staff's questions and positions concerning CESSAR instrument and controls design.
September 24, 1974	Letter from Combustion Engineering regarding steam generator iodine carry over.
September 30, 1974	Submittal of Amendment 7 to CESSAR consisting of partial responses to the staff's August 14 and 22, 1974 requests for additional information.
October 7, 1974	Submittal of Amendment 8 to CESSAR consisting of partial response to the staff's August 14 and 22, 1974 requests for additional information.
October 14, 1974	Submittal of Amendment 9 to CESSAR consisting of partial response to the staff's August 14, 22, and 30, 1974 requests for additional information.
October 15, 1974	Letter from Combustion Engineering providing schedule information concerning the submittal of responses to the staff's requests for additional information.
October 21, 1974	Submittal of Amendment 10 to CESSAR consisting of partial responses to the staff's August 14, and 22, 1974 requests for additional information.
October 22, 1974	Meeting with Combustion Engineering to discuss its comments concerning a standard nuclear steam supply system definition that is being considered by the staff for the CESSAR standard plant, and to discuss the treatment of the current CESSAR optional systems when the standard nuclear steam supply system definition is finalized.
October 24, 1974	Meeting with Combustion Engineering to discuss CESSAR responses in areas reviewed by the Systems Analysis Branch.
October 25, 1974	Letter to Combustion Engineering requesting additional information required as a result of the staff's review of CESSAR Amendments.
October 28, 1974	Submittal of Amendment 11 to CESSAR consisting of partial responses to the staff's August 14 and 22, 1975 requests for additional information.
October 29, 1974	Meeting with Combustion Engineering to discuss unresolved Reactor Systems Branch CESSAR review matters.

712 346

October 30, 1974	Meeting with Combustion Engineering to discuss the design of the steam generator in areas that are reviewed by the Reactor Systems Analysis Branch.
November 4, 1974	Submittal of Amendment 12 to CESSAR consisting of partial responses to the staff's August 14 and 22, 1974 requests for additional information and other changes to CESSAR.
November 5, 1974	Meeting with Combustion Engineering to discuss proposed responses to some of the staff's requests for additional information.
November 11, 1974	Meeting with Combustion Engineering to resolve a substantial list of inadequate CESSAR responses to staff's request for additional information in areas reviewed by the Electrical Instrumentation and Control System Branch.
November 13, 1974	Letter from Combustion Engineering indicating that it intends to submit CESSAR loss-of-coolant accident analysis on January 13, 1975.
November 18, 1974	Submittal of Amendment 13 to CESSAR consisting of partial responses to the staffs August 14 and 22, 1974 requests for additional information.
November 18, 1974	Meeting with Combustion Engineering to discuss the core operating limit supervisory system.
November 19, 1974	Letter to Combustion Engineering requesting additional information on the emergency feedwater system.
November 25, 1974	Submittal of Amendment 14 to CESSAR consisting of additional responses to the staff's August 14 and 22, 1974 and October 25, 1974 requests for additional information.
November 26, 1974	Letter to Combustion Engineering providing a staff position regarding the containment spray system.
November 26, 1974	Letter to Combustion Engineering regarding evaluation of the CESSAR application at a power level other than that recommended by Regulatory Guide 1.49.
December 4, 1974	Meeting with Combustion Engineering to discuss CESSAR interface requirements in the area reviewed by the Auxiliary and Power Conversion System Branch.
December 4, 1974	Meeting with Combustion Engineering to discuss the modifications being made to the CESSAR containment spray system to conform to the staff's position that it must be available (without operator action) whenever the reactor coolant system is above 200 degrees Fahrenheit.
December 16, 1974	Submittal of Amendment 15 consisting of revisions to update and address unresolved CESSAR issues.
December 23, 1974	Submittal of Amendment 16 consisting of revisions to update and address unresolved CESSAR issues.
January 3, 1975	Meeting with Combustion Engineering to discuss outstanding CESSAR review matters.
January 3, 1975	Letter to Combustion Engineering regarding Regulatory Guide 1.49.
January 13, 1975	Submittal of Amendment 17 consisting of revisions to update and address unresolved CESSAR issues.

712 347

January 20, 1975	Submittal of Amendment 18 consisting of revisions to update and address unresolved CESSAR issues.
January 27, 1975	Submittal of Amendment 19 consisting of revisions to update and address unresolved CESSAR issues.
January 31, 1975	Meeting with Combustion Engineering to discuss outstanding items that must be completed by the Combustion Engineering in order for the staff to complete its review of CESSAR and prepare a safety evaluation report.
February 4, 1975	Letter from Combustion Engineering discussing fuel transfer tube closure.
February 10, 1975	Submittal of Amendment 20 consisting of revisions to update and address unresolved CESSAR issues.
February 10-11, 1975	Meeting with Combustion Engineering to discuss the staff's requirements regarding automatic repressurization of the safety injection tanks and safety grade requirements for the circuitry provided to trip the reactor.
February 12, 1975	Letter from Combustion Engineering providing comments on Amendment 1 to WASH-1341.
February 14, 1975	Meeting with Combustion Engineering to discuss its comments on draft copy of the staff's second request for additional information concerning computerized protection system.
February 14, 1975	Letter to Combustion Engineering regarding outstanding review items.
February 18, 1975	Letter to Combustion Engineering requesting additional information regarding the computer protection system.
February 24, 1975	Letter to Combustion Engineering requesting a description of the analysis methods used to derive trip functions.
February 24, 1975	Letter to Combustion Engineering regarding the computer protection system.
February 28, 1975	Letter to Combustion Engineering regarding proprietary status of Appendix 7A to CESSAR.
February 28, 1975 & March 1, 1975	Advisory Committee for Reactor Safeguards Subcommittee meeting on CESSAR.
March 3-4, 1975	Meeting with Combustion Engineering to discuss the computerized protection system design, and to discuss CESSAR interface requirements.
March 3, 1975	Submittal of Amendment 21 consisting of revisions to update and address unresolved CESSAR issues.
March 5, 1975	Meeting with Combustion Engineering to discuss design interface requirements imposed by CESSAR on the balance of plant designs that mate with System 80.
March 7, 1975	Letter from Combustion Engineering providing fuel transfer tube drawings.
March 10, 1975	Submittal of Amendment 22 consisting of revisions to update and address unresolved CESSAR issues.
March 10, 1975	Letter from Combustion Engineering providing the status of outstanding CESSAR review matters.

March 10, 1975	Letter from Combustion Engineering providing information on radioiodine spiking effects on accident releases.
March 10, 1975	Letter from Combustion Engineering indicating that suitable textual changes will be made in the research and development section of CESSAR for core protection calculators.
March 11-12, 1975	Meeting with Combustion Engineering to discuss the CESSAR computerized protection system design.
March 17, 1975	Submittal of Amendment 23 consisting of revisions to update and address unresolved CESSAR issues.
March 21, 1975	Letter to Combustion Engineering in response to its comments on Amendment 1 to WASH-1341.
March 21, 1975	Letter to Combustion Engineering regarding proprietary Appendix 7A to CESSAR.
March 24, 1975	Submittal of Amendment 24 consisting of revisions to update and address unresolved CESSAR issues.
March 31, 1975	Letter from Combustion Engineering regarding proprietary Appendix 7A to CESSAR.
April 3, 1975	Letter to Combustion Engineering regarding consideration of iodine spiking in accident analyses.
April 7, 1975	Submittal of Amendment 25 consisting of revisions to update and address unresolved CESSAR issues.
April 14, 1975	Submittal of Amendment 26 consisting of revisions to update and address unresolved CESSAR issues.
April 23, 1975	Meeting with Combustion Engineering to discuss waste management systems interface requirements.
April 28, 1975	Submittal of Amendment 27 consisting of revisions to update and address unresolved CESSAR issues.
May 1, 1975	Meeting with Combustion Engineering to discuss outstanding CESSAR review matters.
May 5, 1975	Submittal of Amendment 28 consisting of revisions to update and address unresolved CESSAR issues.
May 6, 1975	Meeting with Combustion Engineering to discuss problem areas encountered in the staff's review of auxiliary systems.
May 8, 1975	Meeting with Combustion Engineering to discuss interface requirements.
May 19, 1975	Submittal of Amendment 29 consisting of revisions to address open review items and to provide commitments that were discussed at the May 1, 1975 meeting on outstanding matters.
May 22, 1975	Meeting with Combustion Engineering to discuss interface requirements.
May 22, 1975	Letter from the Advisory Committee on Reactor Safeguards providing comments concerning the CESSAR core protection calculators and core operating limit supervisory system.
May 23, 1975	Advisory Committee on Reactor Safeguards subcommittee meeting to discuss the CESSAR reactor vessel, fuel and core internals.

712 349

May 25, 1975	Letter from Combustion Engineering providing comments on the staffs summary of the May 8, 1975 meeting with the Combustion Engineering.
June 2, 1975	Submittal of Amendment 30 consisting of clarification and discussion of commitments made at the May 1, 1975 meeting with Combustion Engineering on outstanding review matters and discussion of several other outstanding matters that were not the subject of the May 1, 1975 meeting.
June 5, 1975	Meeting with Combustion Engineering to discuss interface requirements.
June 9, 1975	Submittal of Amendment 31 consisting of ECCS performance results for the System 80 design.
June 23, 1975	Submittal of Amendment 32 consisting of clarification and revision of information concerning main steam line interfaces, anticipated transients without scram, and other matters.
June 24, 1975	Letter to Combustion Engineering requesting information concerning the CESSAR design criteria with respect to protection against industrial sabotage.
July 3, 1975	Letter to Advisory Committee on Reactor Safeguards transmitting copies of the Report to the Advisory Committee on Reactor Safeguards in the matter of the Combustion Engineering Standard Safety Analysis Report, dated July 3, 1975.
July 7, 1975	Letter to Combustion Engineering identifying information required by the staff to complete its review of the CESSAR ECCS evaluation.
July 7, 1975	Submittal of Amendment 33 consisting of revised safety injection piping and instrument diagrams that reflect the redesign of the shutdown cooling system.
July 11, 1975	Letter to Combustion Engineering transmitting copies of the Report to the Advisory Committee on Reactor Safeguards in the matter of the Combustion Engineering Standard Safety Analysis Report, dated July 3, 1975.
July 18, 1975	Submittal of Amendment 34 consisting of a revised boron dilution analysis, a response to the industrial security issue, a statement on loose parts monitoring and changes for general updating of CESSAR.
July 21, 1975	Letter from Combustion Engineering providing a schedule for submittal of information required for the staff's evaluation of the ECCS analysis.
July 23, 1975	Letter from Combustion Engineering providing its plans for resolving the outstanding items in the Report to the Advisory Committee on Reactor Safeguards concerning CESSAR.
July 28, 1975	Submittal of Amendment 35 consisting of improvements in the design of the containment hydrogen recombiner system.
August 4, 1975	Submittal of Amendment 36 consisting of modified interface requirements and removing all optional systems for the CESSAR design scope.
August 8, 1975	Issuance of Supplement No. 1 to the Report to the Advisory Committee on Reactor Safeguards.

August 8, 1975	Letter to Combustion Engineering transmitting Supplement No. 1 to Report to Advisory Committee on Reactor Safeguards.
August 11, 1975	Submittal of Amendment 37 addressing diversity of containment isolation, grid frequency ramp down, and independence of plant protection system cabinents which were identified as outstanding items in the Report to the Advisory Committee on Reactor Safeguards.
August 14, 1975	Advisory Committee on Reactor Safeguards meeting to discuss CESSAR.
August 25, 1975	Submittal of Amendment 38 addressing staff comments on the interfaces submitted in Amendment 36.
August 26, 1975	Letter to Combustion Engineering Granting Withholding of Amendment 9.
September 5, 1975	Meeting with Combustion Engineering to discuss Combustion Engineering Electrical submittals through Amendment No. 37 and status of open items.
September 15, 1975	Meeting with Combustion Engineering where they will appeal Electrical, Instrumentation and Control related staff positions concerning loss-of-load bypass and steam generator isolation following a steam line break accident.
September 17, 1975	Letter from the Advisory Committee on Reactor Safeguards on CESSAR.
September 22, 1975	Submittal of Amendment 29 addressing staff comments on interface submitted in Amendment 36.
September 26, 1975	Letter to Combustion Engineering regarding a change in assignment of project managers from G. Rivenbark to I. Villalva.
September 29, 1975	Letter to Combustion Engineering requesting additional information.
October 2, 1975	Meeting with Combustion Engineering to afford them an opportunity to appeal at the Director Level, a Staff position concerning a design proposed by Combustion Engineering which would bypass reactor trip upon certain loss-of-load conditions.
October 6, 1975	Submittal of Amendment 40 providing responses to four outstanding items identified in the Report to the Advisory Committee on Reactor Safeguards.
October 10, 1975	Letter to Combustion Engineering transmitting list of outstanding items on CESSAR.
October 20, 1975	Submittal of Amendment 41 completing responses to all outstanding items identified in Section 1.9 of the Report to the Advisory Committee on Reactor Safeguards.
October 30, 1975	Letter to Combustion Engineering requesting additional information pertaining to reactor vessel head drop analysis.
November 20, 1975	Letter from Combustion Engineering regarding loss-of-load trip and the loss-of-load trip inhibit.
November 25, 1975	Letter to Combustion Engineering regarding pressure vessel support system.
November 26, 1975	Submittal of Amendment 42 provides responses to outstanding issues concerning reactor vessel head drop and long term cooling, additional information on ATWS, balance of plant design, audit intervals for Combustion Engineering vendors, and the loss-of-load trip.

December 2, 1975	Letter from Combustion Engineering in response to our letter on reactor pressure vessel support system.
December 5, 1975	Letter from Combustion Engineering providing further information regarding the boron dilution accident and turbine admission valve size assumptions and reactor vessel head drop.
December 17, 1975	Submittal of Amendment 43 incorporates responses to staff's concerns regarding boron dilution, turbine admission valve size considerations, reactor pressure vessel head drop, and asymmetric loadings of reactor vessel supports.
December 22, 1975	Submittal of Amendment 44 incorporates design modifications to provide independence for engineered safety features actuation system power supplies, provides environmental testing commitment, and testing for reactor coolant pumps without cooling water.

APPENDIX B

BIBLIOGRAPHY

NOTE: Documents referenced in or used to prepare this report, excluding those listed in the CESSAR, may be obtained at the source stated in the bibliography or, where no specified source is given, at most major public libraries. Documents submitted by Combustion Engineering and Commission Rules and Regulations and Regulatory Guides may be inspected at the Commission's Public Document Room, 1717 "H" Street, N. W., Washington, D. C. Specified documents relied upon by the Nuclear Regulatory Commission staff and referenced in this report are as follows:

Reactor

(1) "Technical Report on Densification of Combustion Engineering Fuels," USNRC Staff Report, August 19, 1974. From USNRC, Office of Nuclear Reactor Regulation, Washington, D. C.

(2) P. J. Pankanskie, "BUCKLE: An Analytical Computer Code for Calculating Creep Buckling of an Initially Oval Tube," Battelle Report, BNWL-1784, May 1974, from National Technical Information Center, Springfield, Virginia.

(3) Cadwell, W. R., "PDQ-7 Reference Manual," WAPD-TM-678, January 1968, from National Technical Information Center, Springfield, Virginia.

(4) Redfield, J. A., "CHIC-KIN -- A Fortran Program for Intermediate and Fast Transients in a Water Moderated Reactor," WAPD-TM-479, January 1965, from Energy Research and Development Administration, Technical Information Center, Oak Ridge, Tennessee.

(5) Yasinsky, J. B., Natelson, M., and Hageman, L. A., "TWIGL -- A Program to Solve the Two-Dimensional, Two Group, Space-Time Neutron Diffusion Equations with Temperature Feedback," WAPD-TM-743, February 1968, from National Technical Information Center, Springfield, Virginia.

Engineered Safety Features

(6) Allen, A. O., "The Radiation Chemistry of Water and Aqueous Solutions," Van Nostrand Company, 1961.

(7) Coward, H. F., and Jones, G. W., "Limits of Flammability of Gases and Vapors," Bureau of Mines Bulletin 503, 1952.

(8) "FLOOD/MOD002 - A Code to Determine the Core Reflood Rate for a PWR Plant with 2 Core Vessel Outlet Legs and 4 Core Vessel Inlet Legs," Interim Report, Aeroject Nuclear Company, November 2, 1972.

(9) Moody, F. J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Vol. 87, p. 134, Journal of Heat Transfer, February 1965.

(10) Parsly, L. F., "Design onsiderations of Reactor Containment Spray System Part VI," The Heating of Spray Drops in Air-Steam Atmospheres, USAEC Report ORNL-TM-2412, Oak Ridge, Tennessee, January 1970.

Radwaste Systems

(11) "Final Environmental Statement Concerning Proposed Rulemaking Action: Numerical Guides for Design Objectives and Limiting Conditions to Meet the Criterion 'As Low As Practicable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," WASH 1258, July 1973, USNRC, Public Document Room, 1717 "H" Street, N. W., Washington, D. C.

APPENDIX C

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 17, 1975

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Anders:

Subject: COMBUSTION ENGINEERING STANDARD SAFETY ANALYSIS REPORT - CESSAR-80

At its 185th Meeting, September 11-13, 1975, the Advisory Committee on Reactor Safeguards completed its review of the Application of Combustion Engineering, Inc. for a Preliminary Design Approval (PDA) for its Standard Reference System-80, Safety Analysis Report CESSAR-80. Subcommittee meetings were held with representatives of the Applicant, and the Nuclear Regulatory Commission (NRC) Staff in Windsor, Connecticut, on February 28 and March 1, 1975, and in Washington, D. C., on May 23 and July 25, 1975. The full Committee met with representatives of the NRC Staff and the Applicant at its 184th Meeting August 14-16, 1975. The Committee also had the benefit of the documents listed below.

The Reference System-80 design consists of the nuclear steam supply system (NSSS) with a rated core power of 3800 MW(t), the NSSS control system, reactor protection system, engineered safety features actuation system, chemical and volume control system, shutdown cooling system, safety injection system and fuel handling system. Combustion Engineering will provide, at the option of the user, certain other safety-related systems which are outside the scope of the Reference System-80 design. These non-standard systems will be dealt with in the user's Safety Analysis Reports.

The Reference System-80 has been designed for application to an envelope of plant sites which encompasses all sites approved to date for Combustion Engineering NSSS. CESSAR-80 provides seismic response spectra for all major components, and equipment and piping systems, and other information required to ensure that the balance of plant is designed to protect the Reference System-80 from all site-related hazards. Application of the Reference System-80 design will require an evaluation of each site to confirm its acceptability within the CESSAR-80 envelope. For multiple reactor units at a single station, CESSAR-80 requires that each important safety-related item of the Reference System-80 design be provided for each reactor unit.

712 355

CESSAR-80 will provide safety-related interface requirements information essential to the design of the balance of plant consistent with the assumptions used by Combustion Engineering in its accident analyses. Since the utility-applicant is responsible for instituting the quality assurance programs necessary to assure that all safety-related design requirements have been met, the Committee will review these matters in more detail with the utility-applicants on a case-by-case basis. The Committee recommends that, during the design, procurement, construction, and startup, timely and appropriate interdisciplinary system analyses be carried out to assure complete functional compatibility across each interface for an entire spectrum of anticipated operations and postulated design basis accident conditions.

The NRC Staff has identified several outstanding issues which will require resolution before the issuance of the PDA. The Committee recommends that these matters be resolved in a manner satisfactory to the Staff. The Committee wishes to be kept informed on the resolution of the following issues:

1. The emergency core cooling system evaluation.

2. The analysis of the effects of anticipated transients without scram.

3. Generic review of the effects of failures of reactor pump lubrication oil and component cooling water supply systems.

The most recent ACRS reports on nuclear power stations utilizing Combustion Engineering NSSS are the December 12, 1974 report on the application to construct the 2570 MW(t). St. Lucie Plant, Unit No. 2 and the June 10, 1975 report on the application to operate the 2570 MW(t). St. Lucie Plant, Unit No. 1. The Committee report on the 3390 MW(t). San Onofre Nuclear Power Generating Station, Units Nos. 2 and 3, selected by the Staff for reactor system design comparison with the Reference System-80 design, was issued July 21, 1972. Generic matters which include possible pump overspeed during a loss of coolant accident, transients associated with inadvertent operation of the emergency core cooling system or chemical and volume control system charging pumps, and analyses of postulated ruptures of the steam generator feed line, should be dealt with appropriately by the Staff. With regard to the rupture accident, the Committee recommends that the Staff perform an independent check on the calculation of steam generator blowdown force effects. It is expected that these items will be resolved in a manner satisfactory to the NRC Staff following the PDA and prior to the Final Design Approval (FDA). During the interim period, the Committee will continue to review these items on a case-by-case basis as well as through other appropriate Subcommittee and full Committee meetings.

The peak linear heat generation rate is reduced to 12.1 kw/ft in order to meet the ECCS final acceptance criteria of Appendix K, 10 CFR 50. The Committee recognizes that conservative restrictions used in the NRC-approved ECCS model and the use of a generalized containment envelope yielding low containment pressures may be factors contributing to the imposed reduction in the permissible linear heat generation rates. The reduced limit imposes restrictions on modes for plant operation and becomes dependent on in-core monitoring systems for verification that limits are not exceeded. The Committee recommends that for a standard reactor of this size, larger safety margins, such as obtainable from higher reflooding rates, should be demonstrated. Programs underway by Combustion Engineering, Inc., include analytical and experimental studies aimed at providing the technical base for ECCS model improvements, as well as studying possible changes involving augmented ECC systems. The Committee believes that these programs constitute a sufficient basis for proceeding at this time and that the demonstration of larger safety margins should be part of the first major revised version of the Reference System-80 design which, as stated by Combustion Engineering, Inc., is likely to be submitted for review in about two years.

The Committee needs to complete its review of the suitability of the new 16 x 16 fuel and modified core reactivity controls of the Reference System-80 design which are now scheduled for initial proof testing at Arkansas Nuclear One, Unit No. 2 and at St. Lucie Plant, Unit No. 2. The Committee also needs to complete its review of the new core protection calculator system and the computer-based core operating limit supervisory system which will be incorporated into the Reference System-80 design in the event they are successfully demonstrated at Arkansas Nuclear One, Unit No. 2. The Committee needs to be assured of the dependability of in-core neutron flux sensors for control of reactors operating at low core power peaking factors. For this purpose the Committee recommends that the Staff and the Applicant continue to gather pertinent information from operating CE reactors. The Committee will continue its review of these matters as appropriate documentation is submitted and the improvements sought can be evaluated.

The Committee recognizes the importance to safety and improved designs of developing computational methods to provide best estimate analyses of LOCA and other postulated accidents. The Committee encourages the Applicant and the NRC Staff to accelerate their efforts to this end. The Committee wishes to be kept informed.

712 357

The CESSAR-80 design should include provisions which anticipate the maintenance, inspection, and operational needs of the plant throughout its service life, including cleaning and decontamination of the primary coolant system, and eventual decommissioning. In particular, the Committee believes that the NRC Staff and Combustion Engineering, Inc., should review methods and procedures for removing accumulations of radioactive contamination whereby maintenance and inspection programs can be more effectively and safely carried out.

The Committee believes that Combustion Engineering and the NRC Staff should continue to review the Reference System-80 for design changes that will further improve protection against sabotage.

The Committee believes that methods that seek to develop reference systems through standardization and through replication need to be coupled with ongoing programs that will permit design changes to reference systems which improve safety and which, when justified, will be implemented in a timely manner. Use of reference systems should lead to more efficient and effective licensing reviews. Programs such as CESSAR-80 will contribute to this process. A transition period will be required in which the Committee will still give attention to the items noted, on a case-by-case basis.

The Committee believes that, subject to the above comments and successful completion of the R&D programs, the Combustion Engineering Reference System-80 design can be successfully engineered to serve as a reference system.

Sincerely yours,

WKerr

William Kerr
Chairman

REFERENCES TO THE CESSAR-80 LETTER:

1. Combustion Engineering Standard Safety Analysis Report for System-80 (CESSAR) with Amendments 1 through 36
2. Report to the Advisory Committee on Reactor Safeguards from the Office of Nuclear Reactor Regulation, dated July, 1975
3. Supplement 1 to the Report to the Advisory Committee on Reactor Safeguards from the Office of Nuclear Reactor Regulation, dated August 8, 1975
4. Letter, dated March 24, 1975, Combustion Engineering, Inc., to DRL concerning information on the fuel transfer tube
5. Letter, dated March 10, 1975, Combustion Engineering, Inc., to DRL concerning radioiodine spiking effects on accident releases
6. Letter, dated January 15, 1975, Combustion Engineering Inc., to DRL concerning views on Anticipated Transients Without Scram

712 359

APPENDIX B

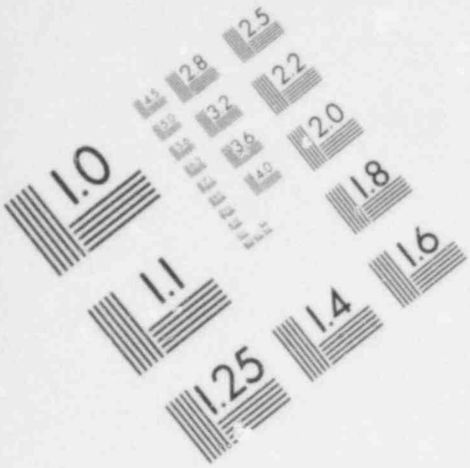
CHRONOLOGY OF REGULATORY RADIOLOGICAL REVIEW

October 10, 1973	Letter to applicant on anticipated transients without scram.
October 25, 1973	Meeting with applicant on quality assurance requirements.
December 28, 1973	Letter from applicant providing information on anticipated transients without scram.
March 1, 1974	Letter from applicant providing topical report on quality assurance program.
March 29, 1974	Application tendered.
April 3, 1974	Letter to applicant acknowledging receipt of application.
May 17, 1974	Meeting with applicant to discuss need for meteorology, hydrology, geology and seismology data.
May 21, 1974	PSAR, general information and antitrust information acceptable for docketing. Environmental report is not acceptable.
May 23, 1974	Application is docketed.
June 3, 1974	Letter to applicant on public document rooms and appeal procedure.
June 17, 1974	Applicant submits revised environmental report and Admendment No. 1 to PSAR.
July 5, 1974	Revised environmental report is acceptable.
August 5, 1974	Letter from applicant provides test boring records.
August 8, 1974	Applicant submits Amendment No. 2 to PSAR.

712 360

August 8, 1974	Environmental report is docketed.
August 29-30, 1974	Site visit and meeting to discuss radiation protection and accident analyses radiological dose calculations.
August 9, 1974	Meeting with applicant to discuss electrical instrumentation and control systems.
August 12, 1974	Applicant submits Amendment No. 3.
August 13-14, 1974	Site visit and meeting to discuss geology, hydrology and seismology.
August 16, 1974	Letter to applicant requesting additional information.
August 29, 1974	Meeting with applicant to discuss responses to requests for additional information.
September 9, 1974	Applicant submits Amendment No. 4.
September 12, 1974	Meeting with applicant to discuss revised construction schedule.
September 14, 1974	Letter to applicant requesting additional information.
September 24, 1974	Letter from applicant on schedule for providing additional information.
September 30, 1974	Applicant submits Amendment No. 5.
October 1, 1974	Letter from applicant providing information on anticipated transients without scram.
October 16-17, 1974	Site visit and meeting to discuss and review meteoro-logical programs.
October 17, 1974	Letter to applicant requesting additional information.
October 11, 1974	Applicant submits Amendment No. 6.
October 25, 1974	Applicant submits Amendment No. 7.

712 361



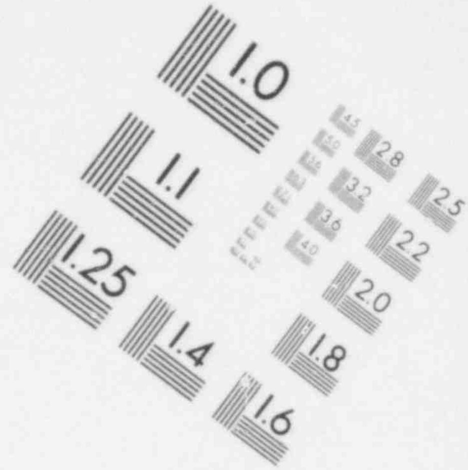
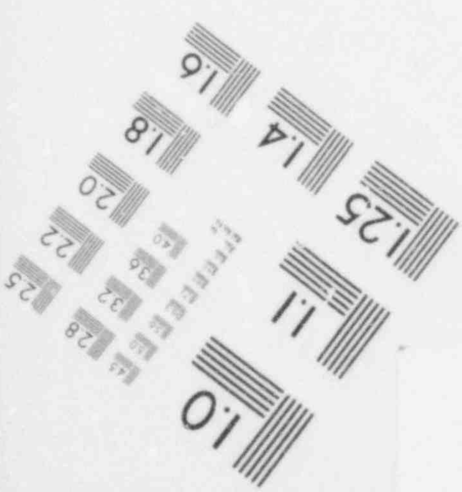
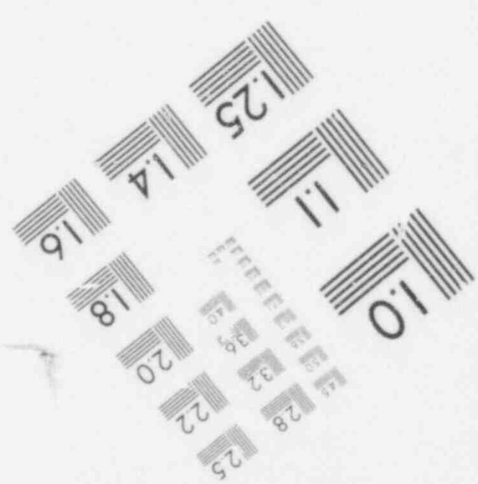
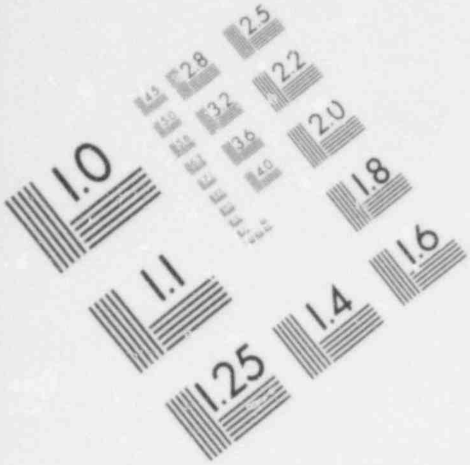


IMAGE EVALUATION
TEST TARGET (MT-3)







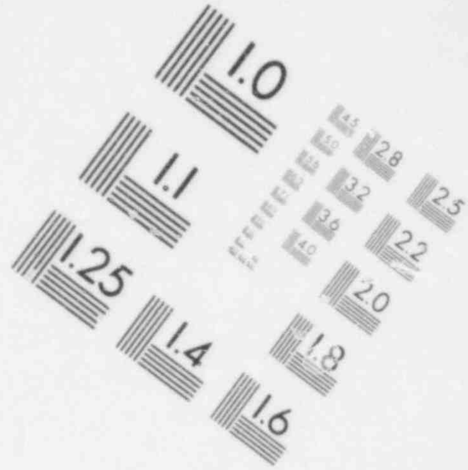
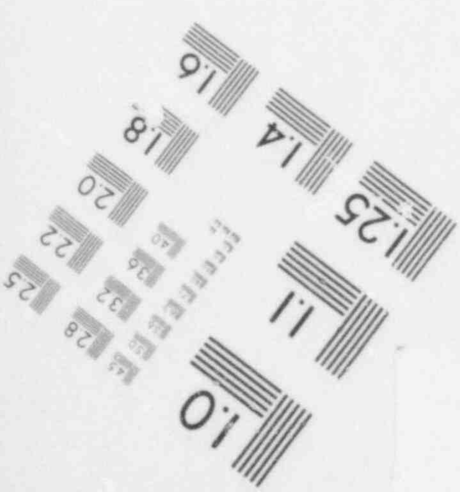
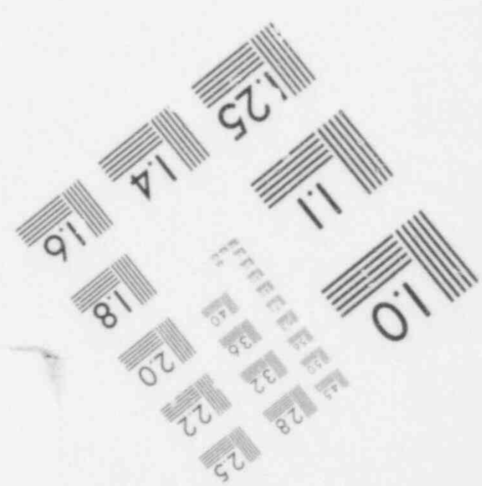


IMAGE EVALUATION TEST TARGET (MT-3)





November 5, 1974	Letter to applicant on the radiological review schedule.
November 7, 1974	Letter to applicant on applicability of new regulatory guides.
November 12, 1974	Site visit and meeting to discuss geology, seismology and foundation engineering information.
November 18, 1974	Applicant submits Amendment No. 8.
November 19-20, 1974	Meeting with applicant to discuss requests for additional information in the areas of structural engineering and auxiliary power and conversion systems.
December 2, 1974	Letter from applicant providing information on aquatic ecology, quantitative community effects and onsite meteorology.
December 9, 1974	Letter to applicant requesting additional information.
December 23, 1974	Applicant submits photomicrographs.
December 23, 1974	Letter to applicant requesting additional information on containment systems.
January 3, 1975	Letter to applicant providing applicant with regulatory position on turbine disk integrity.
January 16, 1975	Letter to applicant requesting additional information on hydrology and geology.
January 27, 1975	Applicant submits Amendment No. 11.
January 27, 1975	Letter from applicant submitting proprietary data in response to letter of December 9, 1974.
February 4, 1975	Meeting with applicant to discuss tornado missiles, containment spray system, leak-chase channels as secondary containment and dewatering and groundwater effects.
February 21, 1975	Applicant submits Amendment No. 12.

~~12 362~~

713 001

February 27, 1975	Meeting with applicant to discuss foundation engineering aspects of dams and dikes for ultimate heat sink and geologic mapping.
March 5, 1975	Letter to applicant requesting additional information on meteorology.
March 12, 1975	Meeting with applicant to discuss industrial security design, types and locations of radiation monitors in ventilation system, design of filters and control room and ventilation.
March 27, 1975	Meeting with applicant to discuss site suitability considerations.
March 27, 1975	Applicant submits Amendment No. 13.
April 8, 1975	Meeting with applicant to discuss the turbine-missile probability study.
April 21, 1975	Applicant submits Amendment No. 14.
April 21, 1975	Applicant submits revision to the industrial security plan.
May 2, 1975	Letter to applicant on anticipated transients without scram and the measures taken to adopt the standard CESSAR design.
May 23, 1975	Applicant submits Amendment No. 15.
June 5, 1975	Meeting with applicant to discuss analyses of containment and associated systems.
June 11, 1975	Applicant submits samples, color photomicrographs and color photographs and a sketch map of the Pee Dee River fault.
July 1, 1975	Applicant submits Amendment No. 16.
July 15, 1975	Applicant submits Amendment No. 17.
July 17, 1975	Meeting with applicant to discuss electrical, instrumentation and control systems.
August 19-20, 1975	Meeting with applicant to discuss issues.

~~712 363~~
713 002

August 22, 1975	Applicant submits Amendment No. 1 to the License Application.
August 22, 1975	Applicant submits Amendment No. 18.
September 5, 1975	Applicant submits Amendment No. 19.
September 17, 1975	Letter from applicant advising that the reactor building auxiliary equipment ventilation system has been modified to conform with position.
September 30, 1975	Letter to applicant requesting additional information on containment.
October 10, 1975	Letter submits Amendment No. 20.
October 10, 1975	Letter from applicant on containment external pressure and stresses.
October 30, 1975	Letter to applicant advising that the proposed secondary containment design is unacceptable.
October 31, 1975	Applicant submits Amendment No. 21.
November 18, 1975	Letter from applicant transmitting outline of annulus ventilation system.
November 28, 1975	Letter to applicant requesting additional information.
December 10, 1975	Assignment of Atomic Safety and Licensing Appeal Board: A. S. Rosenthal, Chairman, Dr. Lawrence R. Quarles, member and Richard S. Salzman, member (published 40 FR 58516, 12/17/75).
December 10, 1975	Decision (ALAB-302).
December 10, 1975	Letter from applicant transmitting final report on annulus ventilation system.
December 22, 1975	Letter from applicant transmitting responses to requests for additional information.
December 24, 1975	Applicant's reply to proposed findings of fact and conclusions of law in the form of an initial decision - environmental and site suitability issues.

December 29, 1975	Letter to applicant regarding asymmetric loading on reactor vessel supports.
January 9, 1976	Letter from applicant regarding asymmetric loading on reactor vessel supports.
January 16, 1976	Letter to applicant on review schedule.
January 22, 1976	Letter to applicant requesting additional information.
February 2, 1976	Applicant submits Amendment No. 22.
February 13, 1976	Letter from applicant on responses to requests for additional information.
February 19, 1976	Letter to applicant on unacceptability of proposed design of the dewatering system.
March 1, 1976	Letter from applicant on dewatering system.
March 8, 1976	Applicant submits Amendment No. 23.
April 2, 1976	Letter to applicant requesting additional information.
April 12, 1976	Letter from applicant on responses to requests for additional information.
April 15, 1976	Letter to applicant requesting additional information.
May 17, 1976	Applicant submits Amendment No. 24.
May 21, 1976	Partial decision on environmental and site suitability issues issued by Atomic Safety and Licensing Board.
May 24, 1976	Letter to applicant on additional financial information.
May 28, 1976	Limited Work Authorization (LWA-1) to allow site preparation.
June 3, 1976	Letter from applicant on schedule for completing review.
June 26, 1976	Letter to applicant on anticipated transients without scram.
July 6, 1976	Applicant submits Amendment 25 to PSAR and Amendment 3 to License application.

713 004

August 10, 1976	Meeting with applicant on dewatering system.
August 10, 1976	Meeting with applicant on geologic fault features.
August 26, 1976	Letter to applicant on rule change.
August 26, 1976	Meeting with applicant on dewatering system.
August 31, 1976	Applicant submits Amendment 26.
September 13, 1976	Applicant submits Amendment 27.
September 30, 1976	Letter to applicant on fire protection.
October 21, 1976	Atomic Safety and Licensing Board Order on reconsideration of partial initial decision.
October 29, 1976	Meeting with applicant on safety issues.
November 23, 1976	Applicant submits Amendment 28.
November 23, 1976	Letter from applicant on fire protection.
December 7, 1976	Letter from applicant on generic issues.
December 13, 1976	Meeting with applicant to discuss LWA request.
December 15, 1976	Letter from applicant on commercial operation dates.
December 16, 1976	Letter from applicant furnishing drawings concerning secondary alarm station and primary alarm station.
December 20, 1976	Letter to applicant transmitting errata sheet for fire protection evaluation letter.
January 13, 1977	Public Hearing to consider issuance of amended Work authorization.
January 19, 1977	Amendment 1 to LWR-1 issued to revise two environmental conditions.
February 23, 1977	Amendment 2 to LWA-1 issued to allow excavation for structures.

February 7, 1977	Letter from applicant furnishing response to seven outstanding items.
February 19, 1977	Letter to applicant requesting additional information on secondary containment bypass leakage.
March 4, 1977	Letter from applicant reporting results of evaluation of a fuel handling accident inside containment.

713 006

APPENDIX C

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS-GENERIC MATTERS

The Advisory Committee on Reactor Safeguards (Committee) periodically issues a report listing various generic matters applicable to large light-water reactors. These are items which the Committee and the Commission's staff, while finding present plant designs acceptable, believe have the potential of adding to the overall safety margin of nuclear power plants, and as such should be considered for application to the extent reasonable and practicable as solutions are found, recognizing that such solutions may occur after completion of the plant. This is consistent with our continuing efforts toward reducing still further the already small risk to the public health and safety for nuclear power plants. The most recent such report concerning these generic items was issued to the Commission Chairman Rowden on April 16, 1976 in a letter from Committee Chairman D. Moeller.

The status of staff efforts leading to resolution of all these generic matters is contained in our Status Report on Generic Items periodically transmitted to the Committee. The latest such Status Report is contained in a letter from B. Rusche to M. Bender dated January 31, 1977. The applicant, in a letter dated December 7, 1977, provided responses to generic items listed in the April 16, 1976 report on generic issues by the Committee. Many of the applicant's responses consisted of references to the CESSAR, the PSAR, or Topical Reports.

For many of the items we have provided in this report specific discussions particularizing for the proposed facility the generic status in the Status Report. These items are listed below with the appropriate section numbers of this report where such discussions are to be found. The numbering corresponds to that in the April 16, 1976 report of the Committee.

For those items applicable to the proposed facility which have not progressed to where specific action can be initiated relevant to individual plants, our Status Report on Generic Items referred to above provides the appropriate information.

Group II

1. Turbine Missiles-Section 3.5.1

2. Effective Operation of Containment Sprays in a LOCA - Sections 6.2.2 and 6.2.8

3. Instruments to Detect Fuel Failures - Section 9.3.1 of Appendix A

5. Monitoring for Excessive Vibration or Loose Parts Inside the Pressure Vessel - Section 5.2.9 and Section 5.2.9 of Appendix A

6. Common Mode Failures - Section 15.6 of Appendix A

Group IIA

1. Pressure in Containment Following a LOCA - Section 6.2.1

4. Rupture of High Pressure Lines Outside Containment - Section 3.6.2

5. Pump Performance During a LOCA - Section 5.2.2

6. Isolation of Low Pressure from High Pressure Systems - Sections 7.6.1 and Section 7.6.3 of Appendix A.

7. Steam Generator Tube Failures - Section 5.5.2 and Section 5.5.2 of Appendix A

Group IIB

1. Hybrid Reactor Protection System - Section 7.2 and Section 7.2.1 of Appendix A

2. Qualification of New Fuel Geometries - Section 1.4 of Appendix A

Group IIC

1. Locking Out of ECCS Power-Operated Valves - Section 7.1.2 and Section 7.1 of Appendix A

2. Fire Protection - Section 9.5.1

3. Design Features to Control Sabotage - Section 13.7

5. Vessel Support Structures - Section 3.9.1.5 of Appendix A

6. Water Hammer - Section 10.5.2 and Section 10.4 of Appendix A

7. Maintenance and Inspection of Plants - Section 12.1

APPENDIX D

BIBLIOGRAPHY

NOTE: Documents referenced in or used to prepare this Safety Evaluation Report, excluding those listed in the Preliminary Safety Analysis Report, may be obtained at the source stated in the Bibliography or, where no specific source is given, at most major public libraries. Correspondence between the Commission and the applicant (Preliminary Safety Analysis Report, Environmental Report, and application) and Commission Rules and Regulations and Regulatory Guides may be inspected at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. Correspondence between the Commission and the applicant may also be inspected at the Public Document Room identified in Section 1.1 of this report. Specific documents relied upon by the Commission's staff and referenced in this Safety Evaluation Report are listed as follows:

METEOROLOGY

1. American Meteorological Society: Hurricane Season Summaries from Weatherwise, published through February 1975; Weatherwise, Inc., Princeton, N. J.

2. Cry, G. W., 1965: Topical Cyclones of the North Atlantic Ocean. Technical Paper No. 55, U. S. Department of Commerce, Weather Bureau. Washington, D. C.

3. Gross, E., 1970: The National Air Pollution Potential Forecast Program. ESSA Technical Memorandum WBTM NMC 47, National Meteorological Center, Washington, D. C.

4. Korshover, J., 1971: Climatology of Stagnating Anticyclones East of the Rocky Mountains, 1936 - 1970. NOAA Technical Memorandum ERL ARL-34, Silver Spring, Maryland.

5. Sagendorf, J. F., 1974: A Program for Evaluating Atmospheric Dispersion from a Nuclear Power Station. NOAA Technical Memorandum ERL ARL-42. Air Resources Laboratory, NAA, Idaho Falls, Idaho.

6. SELS Unit Staff, National Severe Storm Forecast Center 1969: Severe Local Storm Occurrences, 1955 - 1967. ESSA Technical Memorandum WBTM FCST. 12, Office of Meteorological Operations, Silver Spring, Md.

7. Thom, H. C. S., 1963: Tornado Probabilities. Monthly Weather Review, October-December 1963, pp. 730-737.

8. Thom, H. C. S., 1968: New Distributions of Extreme Winds in the United States. Journal of the Structural Division, Proceedings of the American Society of Civil Engineers - July 1968, pp. 1787-1801.

713 009

9. U. S. Department of Commerce, Environmental Data Service: Climatic Atlas of the United States. Environmental Science Services Administration, Washington, D. C. June 1968.

10. U. S. Department of Commerce, Environmental Data Service: Local Climatological Data, Annual Summary with Comparative Data-Charlotte, N. C. Published annually through 1974.

11. U. S. Department of Commerce, Environment-l Data Service: Storm Data Published monthly, Asheville, N. C.

12. U. S. Department of Commerce, Environmental Data Service: Topical Storm and Atlantic Hurricane Articles from the Monthly Weather Review; Monthly Weather Review published through December 1973.

13. U. S. Department of Commerce, Weather Bureau, 1965: Climatograhy of the United States No. 86-27, Supplement for 1951 through 1960, North Carolina. Environmantal Data Service, Asheville, N. C.

GEOLOGY AND SEISMOLOGY

14. Brazee, R. J. (1972) "Attenuation of Modified Mercalli Intensities with Distance for the United States East of 106°W," Earthquake Notes Vol. XLIII, No. 1, pp. 41-52.

15. Coulter, H.W., Waldron, H. H., and Devine, J. F., (1973) "Seismic and Geologic Siting Considerations for Nuclear Facilities," Fifth World Conference on Earthquake Engineering, Rome, paper 302.

16. Hofmann, R. B. (1974), "Factors for Assessing Earthquake Hazards in the United States," Misc. Paper S-73-1, Report 3, U. S. Army Waterways Experiment Station, Vicksburg, Mississippi, 93 pages.

17. Newman, Frank (1958), "Damaging Earthquake and Blast Vibrations," The Trend in Engineering, University of Washington, pp. 5-26.

18. Nuttli, O. W., 1973, "The Mississippi Valley Earthquake of 1811 and 1812: Intensities, Ground Motion and Magnitudes." Bulletin of the Seismological Society of America, Vol. 63, No. 1, pp 227 - 248.

19. Richter, C. F. 1958, "Elementary Seismology." W. H. Freeman and Company - San Francisco, 768 pages.

20. Schnabel, P. B. and Seed, H. B., 1973, "Acceleration in Rock for Earthquakes in the Western United States," Bulletin of the Seismological Society of America, Vol. 63, No. 2.

21. Seed, H. B. and Idriss, I. M., 1969, "Influence of Soil Conditions on Ground Motions During Earthquakes," Journal of Soil Mechanics and Foundations Division, Proceedings, American Society, Civil Engineers.

22. Trifunac, M. D. and Brady, A. G., 1975, "On the Correlation of Seismic Intensity Scales with the Peaks of Recorded Strong Ground Motion." Bulletin of the Seismological Society of America, Vol. 65, No. 1, pp. 139-162.

STRUCTURAL ENGINEERING

23. ASCE Paper No. 3269, "Wind Forces on Structures," Proceedings of the American Society of Civil Engineers, 1961.

24. A. Amirikian, "Design of Protective Structures," Bureau of Yards and Docks, Publication No. NAVDOCK P-51, Department of the Navy, Washington, D. C., August 1950.

25. W. B. Cottrell and A. W. Savolainen, "U. S. Reactor Containment Technology," ORNL-NSIC-5, Vol. 1, Chapter 6.

26. Williamson and Alvy, "Impact Effect of Fragments Striking Structural Elements," Holmes and Harver, Inc., November 197[illegible].

27. American Concrete Institute, "Building Code Requirements for Reinforced Concrete (ACI 318-1971)," P. O. Box 4754, Redford Station, Detroit, Michigan 48219.

28. American Institute of Steel Construction, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," 101 Park Avenue, New York, N. Y. 10017, 1969.

713 011

APPENDIX E

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket Nos. 50-491
Duke Power Company	)	50-492
	)	50-493
(Cherokee Nuclear Station, Units 1, 2 and 3	)	

NRC STAFF EVALUATION OF LIQUID
AND GASEOUS EFFLUENTS WITH RESPECT TO
APPENDIX I OF 10 CFR PART 50

Introduction

On February 20, 1976, the NRC Staff (Staff) received an inquiry from the Atomic Safety and Licensing Board regarding the state of the environmental evaluation of the Duke Power Company's Cherokee Nuclear Station, Units 1, 2 and 3 upon compliance with Appendix I to 10 CFR Part 50. The Board considers the Staff evaluation inadequate with respect to individual doses. The FES indicated that the Staff was in the process of reassessing the parameters and mathematical models and that a detailed assessment to determine conformance with Appendix I would be completed in connection with the hearing on radiological safety aspects of the facility. The purpose of this testimony is to present the results of that detailed assessment. The assessment was performed to determine if the proposed Cherokee Nuclear Station, Units 1, 2 and 3 met the numerical design objectives specified in Sections IIA, B, C and D of Appendix I of 10 CFR Part 50.(1)

713 012

On September 4, 1975[2], the Commission amended Appendix I of 10 CFR Part 50 to provide persons who have filed applications for construction permits for light-water-cooled nuclear power reactors which were docketed on or after January 2, 1971, and prior to June 4, 1976, the option of dispensing with the cost-benefit analysis required by Paragraph II.D of Appendix I. This option permits an applicant to design his radwaste management systems to satisfy the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket RM-50-2, dated February 20, 1974.[3] As indicated in the Statement of Considerations included with the amendment, the Commission noted it is unlikely that further reductions to radioactive material releases would be warranted on a cost-benefit basis for light-water-cooled nuclear power reactors having radwaste systems and equipment determined to be acceptable under the proposed Staff design objectives set forth in RM-50-2.

In Amendment 4 to the Environmental Report (ER), dated October 13, 1975[4], Duke Power Company chose to comply with the Commission's September 4, 1975 amendment to Appendix I, eliminating the necessity to perform a cost-benefit analysis as required by Paragraph II.D of Appendix I.

Evaluation

The Staff has evaluated the radioactive waste management systems proposed for Cherokee Nuclear Station, Units 1, 2 and 3, to reduce the quantities of radioactive materials released to the environment in liquid and gaseous effluents. These systems have been previously described in Section 3.5 of the Final Environmental Statement (FES), dated October 1975.(5) Based on information provided by the applicant in the referenced Amendment 4 to the ER on more recent operating data applicable to the Cherokee Nuclear Station, and on changes in our calculational model, we have generated new liquid and gaseous source terms to determine conformance with Appendix I. These values are different from those given in Tables 3.4 and 3.5 of the FES for Units 1, 2 and 3.

The new source terms, shown in Tables 1 and 2, were calculated using the models and methodology described in Regulatory Guide 1.BB, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors (PWRs)," September 9, 1975. These source terms were used to calculate the doses as described below. The dispersion of radionuclides in and the deposition of radionuclides from the atmosphere were based on analyses performed by the NRC staff for this evaluation.

The mathematical models used to perform the dose calculations are contained in Regulatory Guide 1.AA.(6)

Included in our analysis are dose evaluations of three effluent categories: 1) pathways associated with liquid effluent releases to the Broad River, 2) noble gases released to the atmosphere, and 3) pathways associated with radioiodines, particulates, carbon-14 and tritium released to the atmosphere.

The dose evaluation of pathways associated with liquid effluents was based on the maximum exposed individual. For the total body dose, the individual is an adult consuming 20 kg/yr of fish harvested in the immediate vicinity of the discharge, and recreational use of the shoreline in the immediate vicinity of the discharge for 10 hr/yr. In terms of body organs, the maximum exposed individual is an infant who consumes 510 ℓ/yr of water from the Union Municipal Water supply located 25 miles down stream of the site.

The dose evaluation of noble gases released to the atmosphere included a calculation of beta and gamma air doses at the site boundary and total body and skin doses at the residence having the highest dose. The maximum air doses at or beyond the site boundary were determined to occur 0.35 mile NE of the facility. The location of maximum total body and skin doses were determined to be at a residence located 0.9 mile North.

The dose evaluation of pathways associated with radioiodine, particulates, carbon-14 and tritium released to the atmosphere was also based on the maximum exposed individual. The maximum individual is an infant who consumes 300 ℓ/yr of milk at the farm located 1.2 miles NE of the site and inhales radionuclides at this location.

Since the Guides on Design Objectives apply to all light-water-cooled reactors at a site, it is necessary to compare the total dose from Units 1, 2 and 3 with the Design Objectives contained in the Concluding Statement of Position of the Regulatory Staff.[3] Table 3 provides a comparison of the calculated doses, with the design objectives of Sections IIA, B and C of Appendix I and the proposed NRC staff design objectives set forth in RM-50-2.

As shown in Table 2, the expected quantity of radioactive materials released in liquid effluents from Units 1, 2 and 3 will be less than 5 Ci/yr/reactor (0.19 Ci/yr/reactor), excluding tritium and dissolved gases, in conformance with the amendment to Section II.D. The liquid effluents released from Units 1, 2 and 3 will not result in an annual dose or dose commitment to the total body or to any organ of an individual, in an unrestricted area from all pathways of exposure, in excess of 5 mrem (Table 3).

Based on the NRC staff's evaluation of the gaseous radwaste management systems, the total quantity of radioactive materials released in gaseous

713 016

effluents from Units 1, 2 and 3 will not result in an annual gamma air dose in excess of 10 mrads and a beta air dose in excess of 20 mrads at every location near ground level, at or beyond the site boundary, which could be occupied by individuals (Table 3). As shown in Table 1, the annual total quantity of iodine-131 released in gaseous effluents will be less than 1 Ci/reactor (0.008 Ci/yr/reactor) in conformance with the amendment to Section II.D and the annual total quantity of radioiodine and radioactive particulates released in gaseous effluents from Units 1, 2 and 3 will not result in an annual dose or dose commitment to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 mrem (Table 3).

Conclusion

The NRC staff evaluation demonstrates that the doses associated with the normal operation of the Cherokee Nuclear Station, Units 1, 2 and 3, meet the design objectives of Sections II.A, II.B and II.C of Appendix I of 10 CFR Part 50, and that the expected quantity of radioactive materials released in liquid and gaseous effluents and the aggregate doses meet the design objectives set forth in RM-50-2.

The NRC staff's evaluation shows that the applicant's proposed design of Units 1, 2 and 3 satisfies the criteria specified in the option provided by the Commission's September 4, 1975 amendment to Appendix I and, therefore, meets the requirements of Section II.D of Appendix I of 10 CFR Part 50.

Based on the NRC staff's evaluation, the proposed liquid and gaseous radwaste management systems for the Cherokee Nuclear Station, Units 1, 2 and 3 meet the criteria given in Appendix I and are therefore, acceptable.

Jacques S. Boegli 3/8/76

Jacques S. Boegli
Senior Nuclear Engineer
Effluent Treatment Systems
Division of Site Safety
and Environmental Analysis
Office of Nuclear Regulation
for the NRC Staff

Sworn and subscribed before me, a Notary Public in and for the County of Montgomery, State of Maryland, this day of March 1976.

Carol J. Durham

Carol J. Durham

My Commission expires July 1, 1978.

References

1. Title 10, CFR Part 50, Appendix I. Federal Register, V. 40, p. 19442, May 5, 1975.

2. Title 10, CFR Part 50, Amendment to Paragraph II.D of Appendix I. Federal Register, V. 40, p. 40918, September 4, 1975.

3. U. S Atomic Energy Commission Concluding Statement of Position of the Regulatory Staff (and its Attachment) - Public Rulemaking Hearing on: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criteria "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactors, Docket No. RM-50-2, Washington, D.C., February 20, 1974.

4. Letter: W. H. Owen, Duke Power Company, to D. R. Muller, U. S. Nuclear Regulatory Commission, ER Amendment 4, October 13, 1975.

5. Staff of the U. S. Nuclear Regulatory Commission. "Final Environmental Statement Related to the Construction of Cherokee Nuclear Station", Units 1, 2 and 3, Docket Nos. 50-491, 50-492 and 50-493, Washington, D.C., October 1975.

6. Staff of the U. S. Nuclear Regulatory Commission. Draft Regulatory Guide 1.AA, "Calculation of Annual Average Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Implementing Appendix I," September 23, 1975.

713 019

TABLE 1

CALCULATED RELEASES OF RADIOACTIVE MATERIAL IN GASEOUS EFFLUENT FROM CHEROKEE NUCLEAR STATION, UNITS 1, 2, AND 3

(Ci/yr/reactor)

Radionuclide	Decay Tanks	Reactor Building	Auxiliary Building	Turbine Building	Condenser Air Ejector	Total
Kr-83	a	a	a	a	a	a
Kr-85m	a	2	2	a	2	6
Kr-85	310	41	1	a	a	[illegible]
Kr-87	a	a	1	a	a	1
Kr-88	a	2	5	a	3	10
Kr-89	a	a	a	a	a	a
Xe-131m	2	41	2	a	1	46
Xe-133m	a	41	4	a	3	48
Xe-133	a	5700	320	a	200	6200
Xe-135m	a	a	a	a	a	a
Xe-135	a	13	8	a	5	26
Xe-137	a	a	a	a	a	a
Xe-138	a	a	1	a	a	1
I-131	a	7.9(-4)[b]	4.2(-3)	1.9(-4)	2.6(-3)	7.8(-3)
I-133	a	4.9(-4)	6.1(-3)	2.7(-4)	3.8(-3)	1.1(-2)
Mn-54	4.5(-3)	9.1(-6)	1.8(-4)	c	c	4.7(-3)
Fe-59	1.5(-3)	3.1(-6)	6(-5)	c	c	1.6(-3)
Co-58	1.5(-2)	3.1(-5)	6(-4)	c	c	1.6(-2)
Co-60	7(-3)	1.4(-5)	2.7(-4)	c	c	7.3(-3)
Sr-89	3.3(-4)	7(-7)	1.3(-5)	c	c	3.4(-4)
Sr-90	6(-5)	1.2(-7)	2.4(-6)	c	c	6.3(-5)
Cs-134	4.5(-3)	9.1(-6)	1.8(-4)	c	c	4.7(-3)
Cs-137	7.5(-3)	1.6(-5)	3(-4)	c	c	7.8(-3)
C-14	7	1	a	a	a	8
H-3	-	-	-	-	-	760
Ar-41	c	25	c	c	c	25

NOTE: "a" appearing in the table indicates release is less than 1.0 Ci/yr for noble gas, 0.001 Ci/yr for iodine.
b = exponential notation: $1(-4) = 1 \times 10^{-4}$
c = less than 1% of total for this nuclide

713 020

TABLE 2

CALCULATED RELEASES OF RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS FROM CHEROKEE NUCLEAR STATION, UNITS 1, 2 AND 3

Nuclide[b]	Ci/yr/reactor
Corrosion & Activation Products	
Cr-51	1(-4)[a]
Mn-54	5.2(-4)
Fe-55	1(-4)
Fe-59	6(-5)
Co-58	2.9(-3)
Co-60	4.5(-3)
Zr-95	7(-4)
Nb-95	1(-3)
Np-239	2(-5)
Fission Products	
Br-83	2(-5)
Rb-86	2(-5)
Sr-89	2(-5)
Mo-99	1.7(-3)
Tc-99m	1.6(-3)
Fission Products (cont'd)	
Ru-103	7(-5)
Rh-106	1.2(-3)
Ag-110m	2.2(-4)
Te-127m	2(-5)
Te-127	2(-5)
Te-129m	7(-5)
Te-129	5(-5)
I-130	9(-5)
Te-131m	3(-5)
I-131	8.9(-2)
Te-132	5.8(-4)
I-132	1.7(-3)
I-133	2.6(-2)
Cs-134	1.8(-2)
I-135	4.6(-3)
Cs-136	2.7(-3)
Cs-137	2(-2)
Ba-137m	7.5(-3)
Ce-144	2.6(-3)
All Others	6(-5)
Total (except H-3)	1.9(-1)
H-3	750

a = Exponential notation: $1(-4) = 1 \times 10^{-4}$

b = Nuclides whose release rates are less than 10^{-5} Ci/yr/reactor are not listed individually but are included in the category "All Others".

713 021

TABLE 3

COMPARISON OF CHEROKEE NUCLEAR STATION, UNITS 1, 2 & 3 WITH APPENDIX I TO 10 CFR PART 50, SECTIONS II.A, II.B AND II.C (MAY 5, 1975)[a] AND SECTION II.D, ANNEX (SEPTEMBER 4, 1975)[b]

Criterion	Appendix I[a] Design Objectives	Annex[b] Design Objectives[c]	Calculated Doses
Liquid Effluents			
Dose to total body from all pathways (Adult)	3 mrem/yr/unit	5 mrem/yr/site	0.11 mrem/yr/unit
Dose to any organ from all pathways (Infant-thyroid)	10 mrem/yr/unit	5 mrem/yr/site	0.31 mrem/yr/unit
Noble Gas Effluents[d]			
Gamma dose in air	10 mrad/yr/unit	10 mrad/yr/site	0.46 mrad/yr/unit
Beta dose in air	20 mrad/yr/unit	20 mrad/yr/site	1.3 mrad/yr/unit
Dose to total body of an individual	5 mrem/yr/unit	5 mrem/yr/site	0.028 mrem/yr/unit
Dose to skin of an individual	15 mrem/yr/unit	15 mrem/yr/site	0.077 mrem/yr/unit
Radioiodines and Other Radionuclides Released to the Atmosphere[e]			
Dose to any organ from all pathways (Infant-thyroid)	15 mrem/yr/unit	15 mrem/yr/site	1.1 mrem/yr/unit

[a] Federal Register V. 40, p. 19442, May 5, 1975.

[b] Federal Register V. 40, p. 40816, September 4, 1975.

[c] Design Objectives given on a site basis. Therefore, these design objectives apply to 3 units at the site.

[d] Limited to noble gases only.

[e] Carbon-14 and Tritium have been added to this category.

713 022

APPENDIX F

CHEROKEE NUCLEAR STATION
Procedure for Investigating and Documenting
Geologic Fault Features

INTRODUCTION:

Extensive studies of fault features characteristic of the region and of the project site area have been made at the Catawba Nuclear Station (Docket Nos. 50-413 and -414) during geologic mapping and at the Cherokee project site in support of the Preliminary Safety Analysis Report for project licensing. These studies[1] establish that fault features occur numerously in a variety of forms and that the occurrence of numerous such features can be anticipated in any large excavation in the region.

PURPOSE:

The purpose of this procedure is to establish a means of utilizing data developed in previous studies to correlate significant characteristics of features occurring in new excavations for safety related structures at the Cherokee site without undue repetition of study if a valid analogy can be made. This procedure also establishes a method and the extent that other fault features with no similarity to previous features studied will be documented, studied, and, where necessary, reported.

SCOPE:

This procedure relates directly to those geologic fault features occurring in excavations required for safety related structure foundations. These features may include brecciated zones containing offsets and any other offset or displaced feature of tectonic origin.

PROCEDURE:

1. Geologic mapping will be conducted as stated in Cherokee PSAR Section 2.5.1.2(9).

2. In the event a feature as described in the scope is discovered, the (field) Geologist will notify the Project Civil Engineer as soon as the feature is discovered.

3. The Project Civil Engineer will hold any structure construction in that area until a determination can be made 1) that the feature is similar to features previously studied by observation and that its relevant characteristics can be determined by correlation to previously studied features or 2) that the feature is not similar to any previously studied feature and requires new investigation.

4. For similar features (described in item 3), the (field) Geologist will so notify the Project Civil Engineer who will release work in the area as soon as

---

(1) Reference: Catawba Nuclear Station - PSAR, Chapter 2, Section 2.5, Appendices 2C and 2E
- "Final Geological Report on Brecciated Zon
Cherokee Nuclear Station - PSAR, Chapter 2, Section 2.5, Appendices 2C and 2E

POOR ORIGINAL

713 023

the (field) Geologist has indexed the feature, documented it by detailed mapping and photographs, and established the feature's similarity to a previously established feature. This similarity will be documented by comparing relevant characteristics to features observed and studied at Catawba Nuclear Station during mapping or in any of the numerous test pits opened at the Cherokee site during subsurface investigative studies. Where this similarity can be established and documented, mapping and project work will continue routinely. A tabular summary will be prepared which indexes each occurrence of a feature and makes specific comparison to a previously studied feature. Documentation will be subject to audit during field inspections by NRC.

5. For features where similarity cannot be established by comparison to features previously studied at the Cherokee or Catawba sites, the Project Civil Engineer will continue to hold work in the area and notify the NRC Project Manager of the discovery. The geologic feature will then be left exposed for ten (10) days for NRC inspection.

5a. Duke with the assistance of Law Engineering Testing Company and/or other consultant will map the feature, develop data, and determine if the feature falls in the sequence of geologic events established and reported in the Cherokee PSAR.

5b. A third party independent geologic consultant will be engaged and will visit the site to examine the feature and examine the data developed by Duke and LETCo and/or other consultant. The (field) Geologist will notify the Project Civil Engineer (when geologic mapping, photography and field data gathering have been completed) that investigations have been completed and documented. The independent consultant will report his findings to the Project Civil Engineer.

5c. If the independent consultant concurs with Duke's conclusions, the Project Civil Engineer will then release the area for project construction activity upon completion of item 5b.

5d. For non-similar features a report will be prepared which shall consist of the following:

Description of the feature including the investigation and description of data obtained

Geologic history

Summary and conclusions

Geologic maps and photographs

Report of findings by Geologic Consultant

This report and other data will be available for NRC review whenever requested.

IWP/gc

POOR ORIGINAL